NRC Comment 10: Additional information is needed to support the conclusion that use of interim treatment measures before the completion of the SWPF is consistent with removal of highly radioactive radionuclides to the maximum extent practical.

Basis: The NRC agrees with the conclusion in Reference 4 that the determination of whether highly radioactive radionuclides have been removed to the maximum extent practical can include a wide variety of considerations. However, it is expected that any factors included in the determination will be supported by a technical basis and, when possible, quantitative comparisons.

For example, although it is stated that risk to the public is reduced by continuing sludge processing at the Defense Waste Processing Facility (DWPF) [4], no information is presented to support the amount of risk reduction achieved by continuing waste processing prior to completion of construction of the SWPF. Furthermore, insufficient information is presented to enable a comparison between the increased risks associated with disposing of Deliquification, Dissolution and Adjustment (DDA) and Actinide Removal Process (ARP)/MCU waste in saltstone with the risks associated with postponing treatment until all of the waste can be treated at the SWPF.

Similarly, although it is stated that it is necessary to treat waste with interim procedures prior to the completion of the SWPF because shutdown of the DWPF due to tank space limitations will be economically impractical, a comparison between the costs of shutting down and restarting the DWPF with the costs of implementing the proposed interim treatment procedures and disposing of higher activity waste in the SDF has not been provided. Although it was estimated that it would cost $1 billion to halt and restart waste processing with the DWPF [4], no basis for that estimate was given.

Path Forward: Provide a detailed cost/benefit analysis supporting a comparison of the proposed alternative with alternative treatment plans. The response should address the quantitative and qualitative costs and benefits of treating waste with the SWPF alone as well as the costs and benefits of treating waste with both the ARP/MCU and the SWPF. The response should include:
1) A comparison between the risks to the general public, workers, and inadvertent intruders associated with the proposed treatment plan and the two alternatives (e.g., treating waste with the SWPF alone or treating waste with the ARP/MCU and SWPF). The response should also include an estimate of the risk the tanks currently pose to the public as well as the number of tank-years of waste storage in old style tanks that would be avoided by treating waste with DDA and ARP/MCU instead of waiting to treat waste with the SWPF (e.g., percent reduction). Consideration should be given to the fact that the wastes that have been proposed to be removed are the lowest activity wastes [4].

2) A comparison of the costs associated with at least three alternatives (i.e., the proposed alternative, treating waste at the SWPF alone, and treating waste with the ARP/MCU and SWPF).

The response should address the costs associated with construction and operation of interim procedures and the costs associated with disposing of a higher activity waste on site, as well as the costs of ceasing and restarting sludge processing.

Additional alternatives, such as slowing down the throughput of the DWPF or creating new interim tank storage, should be considered. The comparison should also consider factors other than economic cost (e.g., schedule) and the factors should be converted into a comparable metric (e.g., cost and risk) to the extent practical.

The analysis should reflect uncertainties in the timing of when sludge processing would need to cease due to lack of tank space and the uncertainty in the availability of the ARP, MCU, and SWPF treatment facilities.

SRS Response: This response evaluates and compares costs and benefits associated with three different cases and demonstrates that the salt waste disposition strategy described in the Draft Section 3116 Determination Salt Waste Disposal Savannah River Site, DOE-WD-2005-001, (WD) is the most cost effective case and the case that will provide the lowest overall risk to the site worker and to the general public. Additional details concerning the evaluation performed in response to this RAI comment can be found in "Cost and Benefit Evaluation for Three Salt Waste Treatment Cases at SRS" (d'Entremont et al. 2005).

The following three interim salt strategy cases were evaluated and compared in this response:

Case #1: Baseline Case - The Interim Salt Treatment Strategy is described in the WD and involves initial salt waste disposition using Deliquification, Dissolution, and Adjustment (DDA) followed by Actinide Removal Process (ARP) and Modular CSSX (Caustic Side Solvent Extraction) Unit (MCU) in combination with DDA until the Salt Waste Processing Facility (SWPF) becomes operational. Once the SWPF facility becomes operational, all salt waste streams will be treated using the SWPF.

Case #2: Limited Interim Processing (LIP) Case – This case does not begin initial salt waste disposition until the ARP and MCU facilities begin operation in 2007. No salt waste is processed using the DDA process. Upon start-up of SWPF, ARP/ MCU operations cease and all salt waste is processed using the SWPF.

Case #3: No Interim Processing (NIP) Case – This case does not begin initial salt waste disposition until the SWPF is ready to begin operation in 2009. No salt waste is processed using the DDA process nor with the ARP/MCU facilities. Using this case, all salt waste is processed using the SWPF.

Note that the Baseline Case is the case that was described in the WD and that the other two cases were requested for evaluation in the RAI. These other two cases evaluated herein, namely LIP and NIP, represent scenarios for evaluation only and do not represent detailed plans that have been accepted by either the DOE or by the facilities involved. When these three cases were evaluated, the following fundamental differences were noted between the cases:

Completion of High Level Waste (HLW) System Operations: The Baseline Case resulted in the completion of HLW System operations in 2019. The salt waste dispositioned to SPF/SDF by DDA and ARP/MCU created compliant tank space in the HLW Tank Farms that permitted Defense Waste Processing Facility (DWPF) operations to continue without interruption and permitted SWPF operations to commence processing at forecast production rates. The LIP Case required 3+ years longer to complete HLW system operations than the Baseline Case. The delay in mission completion resulted from the reduced processing rates through SWPF and DWPF caused by the limited compliant tank space available to prepare the salt and sludge waste streams for processing during initial years of SWPF operation. DWPF production rates are impacted because of the limited compliant tank space prevents sludge washing which is required prior to processing sludge waste at DWPF. The NIP Case required approximately 5+ years longer to complete HLW system operations than the Baseline Case. The causes for the delay in mission completion were the same as those above. However, since the time required to recover adequate compliant tank space was longer for this case as compared to the LIP case, the extension of HLW System operations was longer for the NIP case. Note that for both the LIP Case and the NIP Case, DWPF operations were slowed, but the DWPF was not shut down.

Risk: The doses (exposures) associated with each of the three cases were compared as well as the material/facilities at risk. Dose was further broken down in terms of dose to the facility worker, dose to the public from both ongoing operations and from material dispositioned to the SDF, and dose to the inadvertent intruder from the SDF. In order to appropriately characterize the risks from ongoing operations, the differences between the cases in terms of old style Tank Years and Tank Farm waste disposition rate were also evaluated and expressed in Curie Years. The evaluation showed the following:

Table 10-1 Summary of Dose, Tank Years and Curie Year Impacts

Case	Total Dose - All Workers (rem)	Public Dose (mrem/yr)	Average SDF Intruder Dose (mrem/year)	SDF Dose - All Pathways (mrem/yr)	Old Style Tank Years*	Tank Farm Curie Years**
Baseline	890	0.19	9***	2.3	240	3.7E+09
LIP Case *(change from Baseline)*	1100 *(+24%)*	0.19 *(0.0%)*	0	2.3 *(0.0%)*	300 *(+25%)*	4.7E+09 *(+25%)*
NIP Case *(change from Baseline)*	1200 *(+35%)*	0.19 *(0.0%)*	0	2.3 *(0.0%)*	340 *(+42%)*	5.3E+09 *(+42%)*

* Total number of years all old style tanks are in service, e.g., 20 tanks in service for 2 years = 40 Tank- Years

** Total number of years a curie is in the Tank Farms, e.g., 30 MCi in the tank farm for three years = 90M Curie- Years

*** The baseline intruder dose of 9 mrem/year equates to an increase of only 2.5% over the natural background dose of 360 mrem/year

It can be seen from Table 10-1 that the Baseline Case results in significantly lower worker dose (approximately 200 rem less than LIP Case and 300 rem less than NIP case) and significantly shorter time that radioactive material remains in the old style tanks. Intruder doses are higher for the Baseline Case, but the difference (average 9 mrem/year for the maximum 100 year intruder dose) is not significant when compared to average exposure from natural sources of radiation (360 mrem/year (NCRP 1987)). The LIP and NIP Cases show significant increases from the Baseline Case for worker exposure and time that radioactive material remains in the Tank Farm. Thus, using the Interim Salt Processing Baseline Case provides the lowest risk to facility workers from radiation exposure and the shortest time that radionuclides remain in tanks that do not meet secondary containment requirements.

Financial Cost: The Baseline Case is the most cost effective case. The primary reason that the Baseline Case is the most cost effective is the difference in lifecycle costs associated with extending the HLW system (Tank Farms, DWPF, SWPF, Saltstone Production Facility (SPF), Saltstone Disposal Facility (SDF), etc.) operations by 3+ years for the LIP Case and 5+ years for the NIP Case. This results in an additional cost for operation of approximately $1B and $1.5B respectively (unescalated). Since the sunk costs (costs already incurred for the project that are not recoverable) for MCU/ARP construction are high relative to the total project cost, and since the life cycle costs for the HLW system are much higher than the project construction and D&D (decontamination and decommissioning) costs, the life cycle costs dominate the cost comparison. As a result of the significant differences (approximately one order of magnitude) between the project costs remaining for ARP/MCU and the life cycle cost increases for extending facility lifecycles, the case that results in the shortest life cycle will have the lowest financial cost.

Other aspects of the facility operations that were reviewed as a part of this evaluation included consideration of slowing down DWPF rather than shutting down DWPF due to feed streams (sludge batches) to DWPF being unavailable. The slowdown avoids a shutdown of DWPF and subsequent restart. The evaluation shows that slowing down DWPF is preferred over shutdown from a cost perspective and cost comparisons utilized this basis when DWPF operation was evaluated. For the analysis of both the LIP Case and the NIP Case, DWPF operations are maintained at a reduced level to avoid the cost impacts a shutdown and restart.

Qualitative Discussion

These additional factors were considered in the comparative evaluation of the Baseline Case, the LIP Case and the NIP Case. The evaluation of these factors is described below. This evaluation is qualitative since it was not possible to provide a quantitative evaluation of these factors.

Sensitivity to facility start-up delays: Since the primary influence on cost and risk associated with these cases is life cycle, delays in facility start-up will have a

significant impact on both risk and cost. The evaluation assumes that the dates projected for facility start-up will be achieved and that throughput rates will be as forecasted. Delays in facility start-up and reductions in throughput rates would extend the duration of facility operation with associated increases in cost and a decrease in the rate of risk reductions.

The primary influence on cost and risk associated with these cases is the duration of facility operation. The evaluation assumes that the dates projected for facility start-up will be achieved and that throughput rates will be as forecasted. Delays in facility start-up and reductions in throughput rates would extend the duration of facility operation with associated increases in cost and a decrease in benefits. It should be noted that this extension in facility operation is likely greater than a day for day match with a delay in facility start-up. Delays in facility start-up will result in less tank space available for salt batch and sludge batch preparation. It would take years of operation at reduced rates to recover the "lost" tank space. In the cases analyzed, it took 4+ years after SWPF start-up for the LIP Case and 7+ years for the NIP Case for SWPF to achieve forecast processing rates. Attaining these forecast processing rates was limited by the availability of compliant tank space to prepare salt batches to feed SWPF at a rate of seven million gallons of salt waste solution per year.

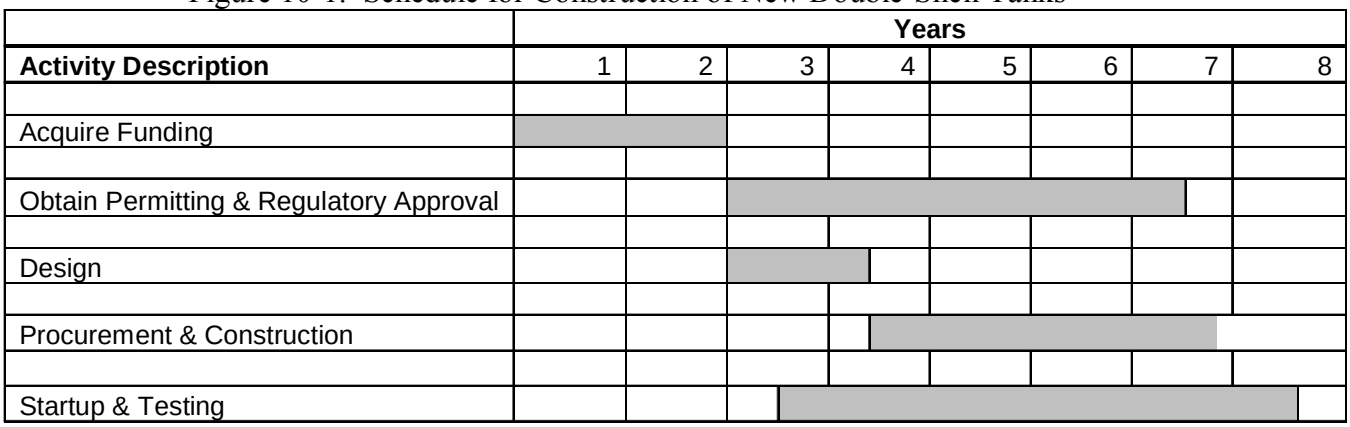
Construction of new HLW storage tanks: In 2001, the cost of new tank construction at Hanford was estimated to be $75 Million assuming that at least four tanks were built (Boyles 2001). The breakdown of the costs supporting this total is shown in Table 10-2. In order to support SWPF start-up at full capacity, four new tanks would need to be constructed for staging dissolved salt solution. Therefore, a total of $300 Million would be required to construct adequate tank space. Since the cost of new tank construction was more than twice the lifecycle cost for ARP/MCU facility (less sunk costs), this was not considered to be cost effective. Hanford also estimated an overall schedule of approximately seven years, the details of which are shown in Figure 10-1. This schedule is not within the timeframe required to support SWPF start-up assumptions. One further note: The construction of new tank space does not support DOE's and the State of South Carolina's overall objective of risk reduction.

Table 10-2. Cost Estimate for Construction of New Double-Shell Tanks (Based on Each One of at Least Four Tanks)

Activity Description	**Cost ($K)**
Obtain Permitting and Regulatory Approval	1,000
Design	7,000
Procurement and Construction	66,000
Start-up and Testing	1,000
Total	**$75,000**

Source: V. C. Boyles, et al. RPP-7702 Tank Space Options, RPP-7702, CH2 M Hill Hanford Group, Rev. 0., April 4, 2001. (page 4-53, Table 4-25)

Figure 10-1. Schedule for Construction of New Double-Shell Tanks



Source: V. C. Boyles, et al. RPP-7702 Tank Space Options, RPP-7702, CH2 M Hill Hanford Group, Rev. 0., April 4, 2001. (page 4-55, Figure 4-8)

Aging Infrastructure: A critical element to the discussion on material at risk (expressed in terms of Curie Years and Tank Years) is the consequence of materials leaked from the aging noncompliant tanks and related infrastructure (pipes, valves, secondary containment structures, etc.). While the sections of Reference 1 that address worker dose and life cycle costs clearly show the expected increases to exposure and cost associated with lifecycle extensions, they cannot accurately quantify the risk associated with the continued use of the aging tank farms during the period of lifecycle extension. Clearly the risk of leaks increases proportionally with the increase in facility lifecycle associated directly with the increase in years of operation, e.g. a ten percent increase in lifecycle is a ten percent increase in risk of an incident. The probability of leaks also increases as a result of the fact that tanks and transfer infrastructure continue to degrade due to the corrosive environment and radiation associated with the storage and processing of HLW. No attempt is made to quantify the probability of failure of the degrading infrastructure, but the increased probability is clear. Likewise, no attempt is made to quantify the impact of the contamination to the environment or to quantify the worker/public dose associated with such a leak. The quantity and type of material, the location of the leak, duration of the leak, proximity of workers, proximity of transport media, environmental conditions, etc. all effect the impact of such an occurrence. While SRS has robust systems for preventing and/or mitigating such an occurrence through tank inspections, corrosion control programs, solution chemistry management, secondary containment, leak detection systems, etc., the probability of occurrence of a leak increases with facility lifecycle extensions. The quantification of Tank Years and curie years is directly related to this increase in risk and demonstrates the exigencies associated with implementation of salt waste stabilization utilizing the Interim Salt Processing Strategy described in the Salt WD.

Summary: Taken as a whole, the above fundamental differences in the cases evaluated demonstrate that the Baseline Case is the most cost effective option and provides the lowest worker dose. Inadvertent intruder doses are marginally higher with the Baseline Case, but this dose is not significant when compared to exposure from natural sources

of radiation. The Baseline Case also reduces radioactive material at risk the most quickly because it facilitates stabilization of radioactive material in the Tank Farms more quickly than in the other cases, as well as permitting closure of old style tanks per the enforceable Federal Facility Agreement (FFA) schedule (WSRC 1993). For these reasons, the Baseline Case provides the greatest overall benefit at the lowest cost.

References:

d'Entremont, P. D., Hill, P. J., Ketusky, E. T., Sheppard, R. E., *Cost and Benefit Evaluation for Three Salt Waste Treatment Cases at SRS*, CBU-PIT-2005-00150, Revision 1, July 7, 2005.

WSRC, 1993, *Federal Facility Agreement for the Savannah River Site*, Administrative Document Number 89005-FF, WSRC-OS-94-42, Effective Date: August 16, 1993.

V. C. Boyles et al., 2001, *RPP-7702 Tank Space Options*, RPP-7702, CH2 M Hill Hanford Group, Rev. 0., April 4, 2001.

National Council on Radiation Protection, 1987, *Ionizing Radiation Exposure of the Population of the United States*, NCRP Report No.93, 1987.

NRC

Comment 11: Predicted removal efficiencies and the bases for predicted removal efficiencies for many of the highly radioactive radionuclides are not provided for each of the treatment schemes (i.e., DDA, ARP, MCU, SWPF). Predicted removal efficiencies and the bases for those removal efficiencies are necessary to support the conclusion that highly radioactive radionuclides have been removed to the maximum extent practical. It should be noted that NRC staff believes that "highly radioactive radionuclides" are those radionuclides that contribute most significantly to risk to the public, workers, and the environment.

Basis: DOE has identified several radionuclides, including I-129, Tc-99, Sn-126, Se-79, Cs-137, Sr-90, Pu-isotopes, U-isotopes, and Np-237/Am-241, as radionuclides that are important to the Saltstone Disposal Facility (SDF) performance [1, 3, 5]. However, the expected removal of all of these radionuclides by the DDA, ARP, MCU, and SWPF treatments are not provided. Predicted removal efficiencies, with the technical bases for the predicted efficiencies, are necessary to support an evaluation of whether the proposed treatment plan is consistent with the removal of highly radioactive radionuclides to the maximum extent practical. Removal efficiencies for unit processes within each of the treatment processes (e.g., cross flow filtration, monosodium titanate (MST) strikes, and solid washing operations) are needed to support the predicted removal efficiencies for each treatment process. Estimated uncertainties in predicted removal efficiencies are necessary to allow a meaningful comparison of the predicted performance of each process and to support an analysis of the source term as part of a performance assessment.

For example, the concentration of several highly radioactive radionuclides in the waste from the SWPF will be higher than the concentrations resulting from the ARP/MCU treatment (Table 3-1 of [5]). Based on the information in Reference 4 and supporting documents, it is difficult to determine if the SWPF waste has higher concentrations of some radionuclides than the ARP/MCU waste because of differences in the predicted radionuclide concentrations in influent waste streams, or because the SWPF will have lower decontamination factors for some radionuclides than the ARP/MCU treatment.

Path Forward: Provide a list of radionuclides that are determined to be highly radioactive radionuclides with respect to waste disposal at the SDF. The response should include technical bases to support the selections. The determination of which radionuclides are highly radioactive with respect to waste disposal at the SDF should address the predicted contributions of each radionuclide to the risk to the public, workers, and the environment under expected conditions and under less favorable conditions (e.g., in cases with significant degradation of the cap, erosion barrier, or waste form).

Provide predicted removal efficiencies for highly radioactive radionuclides for the DDA, ARP, MCU, and SWPF treatment processes, as well as unit processes within each treatment process. The response should include flowcharts showing removal

efficiencies for highly radioactive radionuclides. The response also should include estimated uncertainties in the predicted removal efficiencies.

SRS Response: Based on consultations with the NRC, DOE views "highly radioactive radionuclides" in the context of Section 3116 to be those radionuclides, which, using a risk-informed approach, contribute most significantly to radiological risk to the workers, the public, and the environment. Table 11-1, below, lists these radionuclides for the salt waste at the Savannah River Site. This list takes into account scientific and health physics principles, knowledge and expertise. The scientific rationale for this list is explained in the *Draft Section 3116 Determination* (pp. 29-30) and the additional information discussed below.

Strontium-90, Cs-137, and the alpha-emitting transuranic (TRU) nuclides (alpha-emitting isotopes of Pu, Am, Np and Cm which constitute the majority of the actinides) are the radionuclides for salt waste disposal at Savannah River Site that, on the basis of a risk-informed approach, contribute most significantly to radiological risk to the workers, the public, and the environment. The significance of the contribution of any particular radionuclide to radiological risk and potential dose depends on the concentration and availability of the radionuclide at the time of potential exposure, as recognized by the NRC. See 10 CFR 61.55(a)(1). DOE has compared the risk contributions of the radionuclides to various existing indicators of radiological risk to workers, the public, and the environment. Specifically, the inventories of radionuclides in solidified salt waste (if solidified without use of DDA, ARP/MCU, or SWPF treatment) were compared against NRC Class A concentration limits, and the dose limits for radiation protection in the performance objectives at 10 CFR 61 Subpart C each serving as a quantitative aid in validating which radionuclides are "highly radioactive." In this analytical process, four results are noted[3]. First, note that Sr-90, Cs-137, and the alpha-emitting TRU nuclides (alpha-emitting isotopes of Pu, Am, Np and Cm) are the only radionuclides in this waste that have total inventories in solidified salt waste (if solidified without use of DDA, ARP/MCU, or SWPF treatment) which would result in a dose exceeding the NRC Class A concentration limits (10 CFR 61.55).[4] Second, note that no radionuclides have average inventories in solidified salt waste (if solidified without use of DDA, ARP/MCU, or SWPF treatment) exceeding 10%[5] of the allowable annual public dose of 25 mrem (*See* 10 CFR 61.41)**.** Third, note

[3] For the comparisons to 10 CFR 61.55 Class A concentration limits, 10 CFR 61.41, 10 CFR 61.42 and 10 CFR 61.43, the comparisons were made as if the waste was solidified as grout, using the grout quantity and grout composition which will be used in the SDF, but without first treating the salt waste through DDA, ARP/MCU or SWPF. The solidified waste form was used in this analysis because the solidified waste form will affect the availability of the radionuclides to the environment, human intruder and the public after disposal. As NRC has recognized, the contribution of any particular radionuclide to radiological risk depends on the availability of the radionuclide at the time of exposure as well as its concentration.

[4] Reference to Class A limits is intended only as a tool to assist in screening nuclides for consideration as "highly radioactive." It does not mean that all nuclides that exceed Class A are highly radioactive radionuclides, *per se*.

[5] Use of 10% in this context is not inconsistent with the position adopted by the NRC in another context (decommissioning). Specifically, in that context, the NRC has stated: "NRC staff considers radionuclides and exposure pathways that contribute no greater than 10 percent of the dose criteria to be insignificant contributors" (NUREG 1757, *Consolidated NMSS Decommissioning Guidance*, Vol. 2, Sec. 3.3, p. 3-4).

that Cs-137 is the only radionuclide with an average inventory in solidified salt waste (if solidified without use of DDA, ARP/MCU, or SWPF treatment) in an SDF vault which would result in a dose that exceeds 10% of a allowable annual intruder dose of 500 mrem (*See* 10 CFR 61.42). Fourth, note that Cs-137 is the only radionuclide with a total inventory in solidified salt waste (if solidified without use of DDA, ARP/MCU, or SWPF treatment) which would result in a dose that exceeds 10% of a allowable annual worker gamma dose of 5.0 rem (*See* 10 CFR 61.43). Additionally, note that Sr-90, Cs-137, and the alpha-emitting TRU nuclides are the only radionuclides driving worker inhalation dose.

In summary, comparison of SRS salt characterization data with the four health and environmental indicators given above suggests Sr-90, Cs-137, and the alpha-emitting TRU nuclides (alpha-emitting isotopes of Pu, Am, Np and Cm) are the radionuclides in solidified salt waste (if solidified without use of DDA, ARP/MCU, or SWPF treatment) to be considered highly radioactive radionuclides to be removed to the maximum extent practical (Reboul 2005). Identification of these nuclides is based on the specific facts of this salt waste, and does not necessarily apply to other wastes or to other 3116 waste determinations. DOE's two-phased, three-part salt processing plan provides for removal of these nuclides using a combination of the following five treatment processes: 1) deliquification, dissolution, and adjustment (DDA); 2) actinide removal process (ARP) without monosodium titanate (MST) sorption; 3) ARP with MST sorption; 4) modular caustic side solvent extraction unit (MCU); and 5) Salt Waste Processing Facility (SWPF) treatments. Nominal, lower bounding, and upper bounding removal efficiencies for each of the planned treatment processes are identified in Table 11-1 below (Reboul 2005).

Table 11-1

Treatment Process	Removal Efficiency, %								
	Sr-90			Cs-137			α-emitting TRU		
	Nom	Low	High	Nom	Low	High	Nom	Low	High
DDA	66	46	86	50	30	70	63	43	83
ARP w/o MST	99.6	98.0	99.9	~ 0	~ 0	~ 0	78	50	93
ARP w/ MST	99.997	99.4	99.999	~ 0	~ 0	~ 0	98	90	99.9
MCU	0	0	0	91	90	92	0	0	0
SWPF	99.98	99.4	99.999	99.998	99.99	99.998	96	90	99.5

Selenium-79, Tc-99, Sn-126, I-129, and uranium isotopes were identified in the *Draft Section 3116 Determination,* as having been considered in detail due to their long radiological lives and high potential for mobility in the environment.[6] However, those radionuclides are in such low concentrations in the salt waste that they do not present a significant risk to the workers, the public or the environment. For those contained in

[6] Although discussed in the draft waste determination, these radionuclides may not be those discussed for other waste forms or other sites.

Tables 1 and 2 of 10 CFR 61.55, the radionuclides individually and in combination are well below the concentration limits for Class A waste even if these radionuclides were solidified without treatment (Reboul 2005, Table 2).

Subsequent to the development of the Draft Section 3116 Determination, DOE prepared an updated Special Analysis for the Saltstone Facility (Cook et al. 2005) using improved analytical models and additional sensitivity analyses that more accurately depicted the potential dose impacts of salt waste disposal. This analysis demonstrates that Se-79, Tc-99, Sn-126, I-129, and uranium were found not to exceed any of the indicators discussed above.

The results of the SA as well as conclusions reached using the above analytical process pertaining to radionuclide inventories in the solidified salt waste (if solidified without use of DDA, ARP/MCU, or SWPF treatment) indicate these radionuclides have an insignificant impact on risk and therefore are not necessarily highly radioactive radionuclides for the SRS salt waste. In fact, inventories of all other radionuclides in the solidified salt waste (if solidified without use of DDA, ARP/MCU, or SWPF treatment) are two or more orders of magnitude below the dose-based limits of the performance objectives discussed above (*See* 10 CFR 61.41-43). As an example, Table 11-2 below demonstrates this for Se-79, Tc-99, Sn-126, I-129, and the uranium isotopes for the all-pathways public dose (10 CFR 61.41).

Table 11-2

Radionuclide	Dose from Average Untreated Solidified Inventories in Vault 4 Volume, mrem/yr	Fraction of 25 mrem/yr Dose
Se-79	3.3E-01	1.3E-2
Tc-99	4.5E-13	1.8E-14
Sn-126	3.0E-17	1.2E-18
I-129	6.3E-03	2.5E-4
U-232	< 1.1E-21	< 4.3E-23
U-233	< 1.6E-19	< 6.3E-21
U-234	< 1.1E-19	< 4.2E-21
U-235	< 3.0E-21	< 1.2E-22
U-236	< 1.3E-20	< 5.0E-22
U-238	< 1.7E-19	< 6.9E-21

(Data derived from Reboul 2005, Table 3).

Thus, these other radionuclides (Se-79, Tc-99, Sn-126, I-129 and uranium isotopes) both individually and in combination would result in doses which are clearly below 10% of the limits set forth in performance objectives in 10 CFR 61.41, 10 CFR 61.42 and 10 CFR 61.43 and therefore do not contribute significantly to the risk to workers, the public, and the environment. Because of the very low concentrations of these

radionuclides and low associated risks as shown above, these radionuclides are not targeted for removal. In this regard, the "maximum extent practical" removal standard in Section 3116 of the NDAA contemplates, among other things, the exercise of expert judgment and consideration of the sensibleness and reasonableness of further removal of radionuclides. For the SRS salt waste streams, the concentrations of Se-79, Tc-99, I-129, Sn-126, and uranium isotopes and the associated risks are so low that it would not be sensible or reasonable to further remove those radionuclides. Nevertheless, DOE notes that additional incidental removal of the radionuclides will likely occur.[7]

With respect to the planned treatment processes, all flowcharts and assumptions are provided in CBU-PIT-2005-00141 (Reboul 2005).

Degradation of the waste cap, erosion barrier, and waste form were included in the analysis quantifying the radionuclide inventory limits for Vault 4 of the SDF (Cook et al. 2005). These limits were the bases for evaluating public and intruder doses in determining highly radioactive radionuclides (Reboul 2005). Consequently, the effects of waste degradation have been taken into account.

References:

Reboul, S. H., 2005. *Removal of Highly Radioactive Nuclides from SRS Salt Waste*, CBU-PIT-2005-00141, Westinghouse Savannah River Company, Closure Business Unit, Aiken, South Carolina. June, 2005.

DOE, February 28, 2005. *Draft Section 3116 Determination Salt Waste Disposal Savannah River Site.* DOE-WD-2005-001.

Cook, J. R., Wilhite, E. L., Hiergesell, R. A., and Flach, G. P. 2005. *Special Analysis: Revision of Saltstone Vault 4 Disposal Limits*, WSRC-TR-2005-00074, Revision 0, May 2005.

NRC, September, 2003. *Consolidated NMSS Decommissioning Guidance*, Vol. 2. NUREG 1757.

[7] During the filtration steps at ARP and SWPF, a majority of the insoluble fractions of these radionuclides will be removed. Additional incidental removal occurs during DDA due to settling.

NRC

Comment 12: Additional information about the selection and optimization of treatment steps in the DDA treatment process and the selection of waste for DDA processing is necessary to support the conclusion that highly radioactive radionuclides have been removed to the maximum extent practical.

Basis: Results of both DOE and independent NRC analyses indicate that several radionuclides (e.g., I-129, Tc-99, Sn-126, Se-79, Cs-137, Sr-90, Pu-isotopes, Np-237/Am-241) are important to SDF performance. Significant fractions of the inventory of most of these radionuclides at the SDF will be attributable to the DDA waste [5]. However, processes to minimize the concentration of many of these radionuclides in the DDA waste are not discussed in the waste determination or supporting documents. For example, attempts to minimize the amount of Sn-126 or actinides in DDA waste might include steps to minimize the amount of sludge entrained in the waste during the DDA process; however, the waste determination does not include a description of the variables that affect the amount of sludge that is entrained or any steps that could be taken to minimize the amount of entrained sludge.

Similarly, although the waste determination indicates that settling is expected to remove a "significant portion" of the insoluble radionuclides (pg. 15 of [4]), it is unclear what removal efficiencies are expected, what data there is to support the expected removal efficiencies, and how the process has been optimized. Because the expected removal efficiencies and factors affecting the removal efficiencies are not discussed, it is unclear whether additional treatment steps, such as filtration, would be practical or if currently planned treatment steps, such as settling, could be improved.

In Reference 4 it is indicated that the lowest activity waste will be selected for DDA processing; however, a comparison of the radionuclide concentrations of the wastes prior to processing is not provided.

Path Forward: Provide information to support the conclusion that the lowest activity waste will be selected for processing in the DDA. Provide information about the selection and optimization of treatment steps to minimize the concentration of highly radioactive radionuclides in DDA waste. The response should include a description of:

1) Factors that affect the amount of sludge entrained in the DDA waste, and efforts to optimize the process to minimize the amount of entrained sludge.

2) Alternative deliquification technologies that were evaluated and the expected removal efficiencies of highly radioactive radionuclides by those technologies. The response should address whether any technologies, such as vacuum techniques, that have been employed with some success at other sites (e.g., Hanford) were considered.

This response also should address the potential effects of differences in the porosity and pore structure of saltcake in different tanks and the potential effects of these differences on the success of the deliquification processes.

3) Alternative filtration technologies that were evaluated and the expected removal efficiencies of highly radioactive radionuclides by those technologies.

In addition, a detailed cost-benefit analysis of the alternative treatment technologies should be provided to support a determination of whether the proposed DDA process is consistent with the removal of highly radioactive radionuclides to the maximum extent practical.

SRS Response: The Deliquification, Dissolution, and Adjustment (DDA) process is described in Section 2, pages 12 – 16, of the Draft 3116 Determination [for] Salt Waste Disposal [at the] Savannah River Site (Salt WD). The response to the questions in this NRC Comment are addressed through discussion of the following topics.

- Saltcake composition considered for DDA;
- Technologies explored;
- Parameters important to optimizing DDA treatment of predominantly soluble Radionuclides;
- Parameters important to optimizing DDA treatment of predominantly insoluble Radionuclides;
- Selection of DDA tanks.

Saltcake Composition Considered for DDA

Most of the waste in the Tank Farms was generated from the chemical separation processes in F- and H-Canyons. This waste contained a strongly acidic solution of nitric acid and metal oxides. Before transferring to the Tank Farms, chemicals (sodium hydroxide) were added to adjust the waste to an alkaline state to prevent corrosion of the carbon steel waste tanks. This chemical adjustment resulted in the precipitation of metal oxides, including strontium (Sr) and actinides (e.g., plutonium (Pu)). These solids settled to the bottom of the waste tanks forming a layer that is commonly referred to as sludge. Since the early 1960s, DOE concentrated the decanted supernate with the Tank Farm evaporator systems to reduce the overall volume of the waste. During the evaporation process, the salt waste was concentrated and formed two distinct phases – concentrated supernate solution and solid saltcake.
By decanting the liquid above the sludge layer, the quantity of entrained solids within the salt phases was minimized. The concentrated supernate and interstitial liquid within the saltcake waste contain the soluble fractions of Cs-137, I-129, Tc-99, Sn-126 and Se-79. The relative portions of the specific isotopes of interest, i.e., Cs-137, I-129, Tc-99, Sn-126, and Se-79, in the supernate phase of the salt waste are noted in Table 12-1.

Table 12-1: Salt Waste Radionuclides (Reboul 2005)

Isotope	Total Inventory in Salt Waste (supernate and saltcake), Ci	Portion of Total Salt Waste Inventory in Supernate
Cs-137	1.1E+8	99+%
I-129	1.8E+1	99+%
Tc-99	3.5E+4	94%
Sn-126	6.2E+2	73%
Se-79	2.2E+2	40%

The solid saltcake is composed predominantly of nitrate, carbonate, aluminate, and sulfate salts and contains relatively small quantities of radioactive material (Drumm and Tran 2004). The radioactive constituents within the solid saltcake are determined predominantly by the quantity of entrained insoluble solids that was carried over during the evaporation process. The insoluble solids entrained in the saltcake include strontium and actinides, as well as the insoluble fractions of Tc-99, Sn-126, and Se-79.

As discussed on page 14 of the Salt WD, the DDA process effectively removes approximately 50% of the soluble nuclides, e.g., Cs-137, I-129, Tc-99, Sn-126, and Se-79, during the deliquification phase and insoluble nuclides, e.g., Sr-90, Pu-isotopes, and Np-237/Am-241 during settling before disposal at Saltstone Disposal Facility (SDF). Optimizing the amount of liquid removed during deliquification optimizes the removal of the soluble nuclides. Likewise, minimizing the amount of entrained insoluble solids during dissolution and optimizing the settling step will minimize the amount of Sr and actinides that are disposed of in the SDF. Optimization of both soluble and insoluble activity is discussed in detail later in this response. A discussion on entrained sludge is found in the response to NRC Comment 15. It should be noted that, because DDA removes a large fraction of the soluble radionuclides from the saltcake prior to dissolution, and because the selected treatment processes target the removal of Cs-137, Sr-90, and the actinides, the DDA stream disposed of in the SDF is actually lower in concentration for I-129, Tc-99, Sn-126, and Se-79, than the ARP/MCU or SWPF streams (d'Entremont and Drumm 2005).

Technologies Explored

The salt treatment technologies explored were not restricted to exclusively to the DDA processing steps, but rather included many alternative processes for treating salt wastes. The DDA process has origins as a proposal to supplement or enhance the now defunct In-Tank Precipitation (ITP) process. When ITP was shut down, an equivalent process to DDA, then referred to as Low Curie Salt, was considered as part of more than 150 process alternatives evaluated for technical viability and effectiveness and for cost for replacing the ITP process (WSRC 1998).

The DDA was rejected as a process incapable of treating **all** saltcake wastes. However, the National Research Council of the National Academy of Sciences (NAS)

recommended that SRS consider "tailoring the processing operations to tank waste contents". The NAS recommendation further states (National Research Council 2000):

> "instead of blending tank wastes to produce a feed that might allow all tank contents to be treated by a single process, as is now planned, would it be advantageous to tailor processing based on chemical and radionuclide contents of individual tanks? For example, could tank wastes with little or no cesium be processed only to remove strontium and actinides … ? Alternatively, could tank wastes with low strontium and actinide concentrations be processed only to remove cesium? Indeed, could tank wastes with low actinide, strontium, and cesium concentrations be sent directly to the Saltstone Facility after minor waste conditioning (e.g., filtration)?"

Based on this recommendation, the waste processing strategy changed to include multiple processes to treat the salt waste based on composition. DDA was developed as a viable process for a portion of the salt waste tanks. By recognizing that there are some saltcakes with lower concentrations of highly radioactive radionuclides, decontaminated dissolved salt solutions can be readily produced that meet the process requirements. This is accomplished by removal of the liquid phase of the salt waste containing the soluble nuclides. This liquid is stored in other waste tanks for future processing through SWPF. The low-activity saltcake remaining is then dissolved and transferred to another tank to separate the liquid phase from the solid phase by allowing the solid phase to settle to the tank bottom. Further partitioning of this low-curie content waste by decanting produces a waste stream that can be disposed of prior to the construction of enhanced processing facilities. The result of this effort is that waste tanks can be emptied and closed earlier than originally planned, and thereby expediting the elimination of the risk associated with storing legacy radioactive liquid waste.

An additional evaluation of potential process alternatives was performed in 2003. This evaluation included several hundred variations of options for methods of removal of salt waste from the waste tanks and disposal of the final waste. The treatment processes were primarily the same as those previously considered in reference WSRC 1998, but varied considerably in physical size, form, and potential location of the processes because the intent of this evaluation was to identify any possible process that could create and maintain adequate operating space in the Tank Farms prior to startup of the SWPF. The treatment process technologies considered include ion exchange, solvent extraction, crystallization, vitrification, reformation, precipitation, geological and electrochemical technologies as well as various methods to physically extract saltcake from a waste tank such as robotic mining, vacuum mining, and sluicing. The evaluation supported the currently planned DDA and ARP/MCU interim salt waste disposition processes.

The response to NRC Comment 10 provides an evaluation of the Interim Processing Plan (Baseline Case), and two other cases suggested for comparison by the NRC. These cases were a limited interim processing case where only the ARP/MCU processes were run before SWPF start-up (no DDA processing) and a no interim processing case where no salt waste was processed until SWPF started up (no DDA or ARP/MCU processing). The fundamental differences in the cases evaluated

demonstrate that the Baseline Case is the most cost-effective option and provides the lowest worker dose. Public doses (including inadvertent intruder doses) are marginally higher with the Baseline Case, but this dose is not significant when compared to exposure from natural sources of radiation (360 mrem/year). The Baseline Case also reduces radioactive material at risk the most quickly because it facilitates stabilization of radioactive material in the Tank Farms more quickly than in the other cases, allowing closure of old style tanks per the Federal Facility Agreement (FFA) schedule. For these reasons, the Baseline Case provides the greatest overall benefit at the lowest cost.

Some alternatives to each DDA process step were evaluated in order to optimize the radionuclide removal effectiveness. DDA can be divided into an initial liquid-solid separation step where the liquid phase is separated from the saltcake and a second liquid-solid separation where insoluble solids are separated from the dissolved salt solution. Key to the liquid phase separation from the saltcake is the deliquification step. The development of the deliquification step considered alternative variations to produce the best possible separation of liquid from solid saltcake. Given the large size of the waste tank, hydrogeology principles were applied to determine favorable configurations for the best separation. Evaluations included one or two pump wells, up to four deliquification cycles, liquid injection on top of saltcake (both in one injection well or in two injection wells), and liquid covering saltcake or not covering saltcake (Staheli and Peters 1998). Initial development lacked SRS saltcake specific physical properties, but experience with deliquifying Hanford saltcakes provided some initial design guidance (Kirk 1980, Handy 1975, Simmons 1995). Cost was not considered as a factor for designing the deliquification step.

Several possible unit operations were considered for the separation of dissolved salt solution from the insoluble solids. The options considered include settling, cross-flow filtration, and dead-end filtration (Norton et al. 2003, Seufert and Norton 2003). The set of options considered follows from site experience and past alternative evaluations for filtering sludge slurries (Poirier 2000, Van Pelt 2000, McCabe 1995, Poirier et al. 2001). Settling was selected since significant reduction in insoluble solids results by using this process and any filtering options would require the design and construction of new facilities. These filtration facilities could not be constructed and placed on-line within the time period needed for initiation of salt waste removal. If the filtration alternative must be pursued due to the time constraints associated with settling unit operation cycle time, there will be schedule impact and life-cycle cost impact associated with the programmatic delays. Additional discussion of the lifecycle cost impacts of programmatic delays is provided in the response to NRC Comment 10.

As an example, 50% of the solids can be removed by settling for less than 19 days for a 300-inch deep batch in the settling tank. The Baseline Case for DDA is to settle for a minimum of 30 days which results in the removal of approximately two-thirds of the radioactive solids through settling (Gillam 2005).

Parameters Important to Optimizing DDA Treatment of Predominantly Soluble Radionuclides

In the deliquification step of DDA, the free supernate is removed from the tank and is stored in another Type III tank for future processing at SWPF. Therefore, the radionuclides associated with the free supernate (e.g., Cs-137, I-129, Tc-99, Sn-126, and Se-79) are removed and stored for future processing through the Salt Waste Processing Facility (SWPF). The second part of the deliquification process involves the removal of the interstitial liquid from the saltcake and storage of this liquid for future processing at the SWPF.

The key to success of the DDA process is the quality of the liquid-solid separation, which is determined by the quantity of liquid residual in the saltcake after deliquification. The key properties that determine residual are intrinsic saltcake permeability, pump out rate, and liquid retention curve. Saltcake porosity primarily identifies the total quantity of the starting liquid phase and is not important to the success of the separation. However, the porosity will affect the total time required to complete the deliquification step.

The intrinsic permeability affects the rate of liquid removal and, thus, the amount of time required to remove the liquid from the saltcake. The higher the permeability, the less time required; the lower the permeability, the greater the time required. If permeability is so low that effective liquid removal would take several years, the process would be impractical to implement.

Information about the potential effects of variations in saltcake properties is continuing to be developed with each SRS waste tank deliquified. When SRS began development of the DDA process, no physical property data on real SRS saltcake was available. Considerable data on Hanford saltcakes was available from their experiences with performing saltcake deliquification operations since the 1970s (Kirk 1980, Handy 1975). Data from simulated SRS saltcake was available for comparison (Wiersma 1996, Kiser 1979, Churnetski 1981, Goodlett 1968). As such, the initial development included simulations that included a large range of variability and heterogeneity in saltcake properties to determine the magnitude of the effect.

The initial range of permeability analyzed included 1.0E-3 to 1.0E-7 cm/sec, a very high to very low permeability and a variation over 10,000 times the lowest value. These initial simulations started with 22 volume % interstitial liquid and resulted in a residual liquid volume from 6.4 – 11% of total saltcake volume after 1000 hours for a single deliquification cycle (Staheli and Peters 1998).

The same initial analysis considered the effect of refilling the pores with a radiologically “clean” liquid and repeating the deliquification cycle. However, refilling the pores by this method would require a substantial addition of clean materials and increase the volume of material to treat for disposal (Staheli and Peters 1998).

In addition, to be successful, this added material would need to be chemically similar to the liquid removed during the first deliquification cycle in order to avoid dissolving saltcake in the process. For example, one million gallons of saltcake would require a refill volume of approximately 250,000 gallons with subsequent storage of the removed liquid. Thus, no advantage is created by any additional deliquification cycles at the cost of the creation of additional waste volume (Staheli and Peters 1998).

The key property to the quality of separation of the liquid phase from the solid saltcake is the amount of residual liquid in the saltcake after deliquification. The residual liquid is an inherent property of the saltcake and varies with the physical structure of the saltcake crystals as well as the chemistry of the liquid and solids. This property is described by hydraulic liquid retention curves because the actual residual content varies with elevation within the saltcake. With empirical data obtained from the first deliquification operation on Tank 41 saltcake in 2002 – 2003, an appropriate range of liquid retention curves was identified (Flach 2003). SRS completed an analysis of deliquification of saltcake with variations in liquid retention curves and initial liquid content (Barnes and Flach 2005, Pike 2005).

In addition to the nominal case of the best-estimated property values, a few select cases were simulated that represent known variability in the properties. These case runs provide an indication of what could reasonably be expected from variability already known to exist. The initial liquid content was nominally determined to be 30 volume % of the saltcake based on data from Tank 41. The analysis included variation of initial liquid content from 25 to 40 volume % of the saltcake. The simulation ran until the removal rate reached 1 gpm average or about 500 to 700 hours of deliquification. The very lowest residual possible for the range of liquid retention curves is 10 to 15% of the saltcake volume (Pike 2005).

Figure 12-1 and Figure 12-2 provide the cases included in the analysis and the relationships between each possible variation. These figures also show the key results of each case. The cases were split into two groups that varied three parameters in order to simplify interpretation. The cube represents the three parameters varied between each set of cases. The axis for each dimension of the cube represents the range of variation expected or known for each parameter. The orientation of the range of values, i.e., high to low, was arranged such that the bottom front left corner represents the least aggressive, least favorable property combination. This combination would be expected to be the least residual liquid, the slowest rate, and, perhaps, the least volume removed. The upper back right corner represents the most aggressive, most favorable property combination.

The results depicted in Figure 12-1 and Figure 12-2 show comparable hydraulically equivalent endpoints, i.e., equivalent hydraulic pressure. The results depicted in the figures were pulled from the case runs when approximately 1 gpm interstitial liquid flow rate is achieved. Continuing to remove liquid in any case will produce a lower residual liquid in the saltcake, but this part of the removal curve also represents the least productive portion of the operation.

Figure 12-3 shows how the continued removal of liquid produces increasingly diminishing progress. Considering that the best process outcome is the lowest amount of residual liquid, not necessarily the most removed liquid, the figures show both values along with estimated time to reach the end state.

Figure 12-1 shows that the time to reach the end state changes relatively little compared to the dramatic variation in removed and residual volumes. Two of the three axis parameters, well height and temperature, can be controlled to some degree by the design of the operation. The variation in intrinsic permeability results in the most variation. Notice that the case with the least residual will take longer and produce more removed liquid volume even though the same stopping point is achieved.

Figure 12-2 shows the variation caused by properties that change the initial liquid volume and volume of retained residual. The cases 1 – 7 were chosen as most physically likely. The analyst considered the combination represented by case 8 and 9 as unlikely combinations that could not readily exist. Cases 8 and 9 were run to make this summary more complete. This figure shows that there is considerably more variability inherent in saltcake physical properties than in controlled properties.

Analysis of the deliquification experience of SRS saltcakes up to now show that approximately 50 percent of the initial volume of liquid can readily be separated from the saltcake. By allowing deliquification to reach infinite duration and allowing for the most favorable variation in physical properties, the most liquid that can be removed is roughly 65% of the original liquid in the saltcake. Deliquification step is considered complete once the average liquid removal rate falls below 1 gpm.

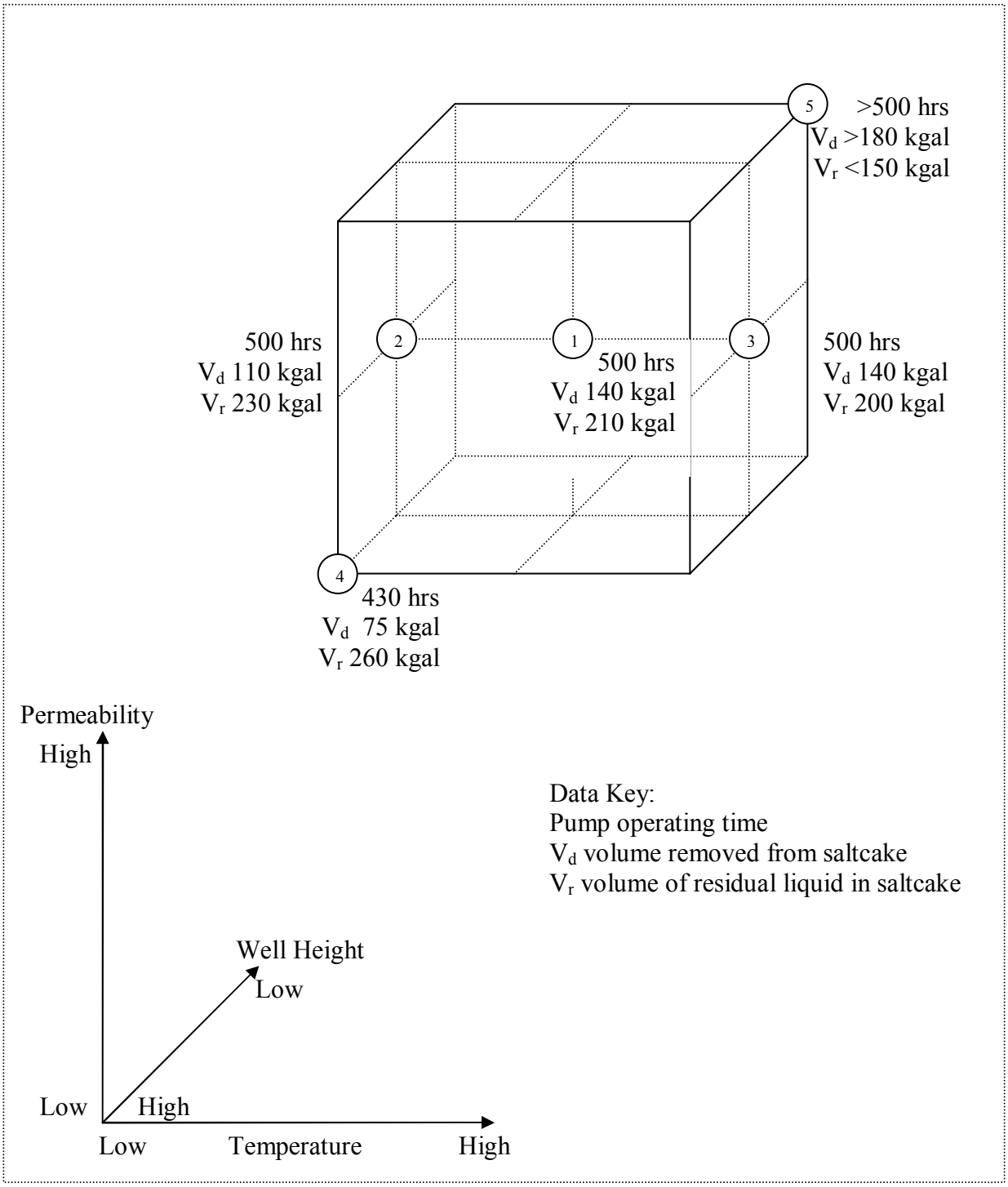


Figure 12-1: Representation of Cases 1 Through 5

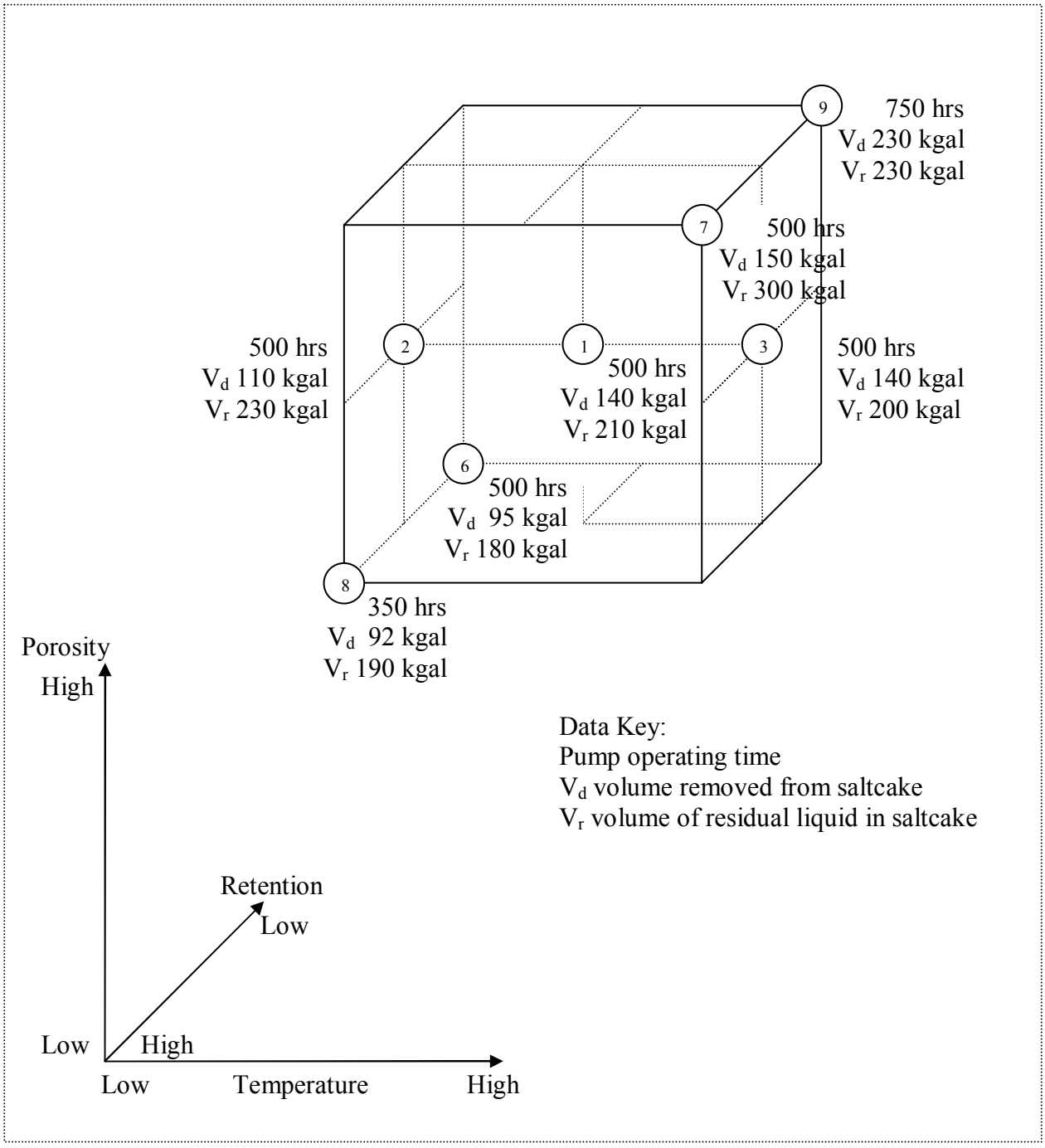


Figure 12-2: Representation of Cases 6 Through 9 and 1 Through 3

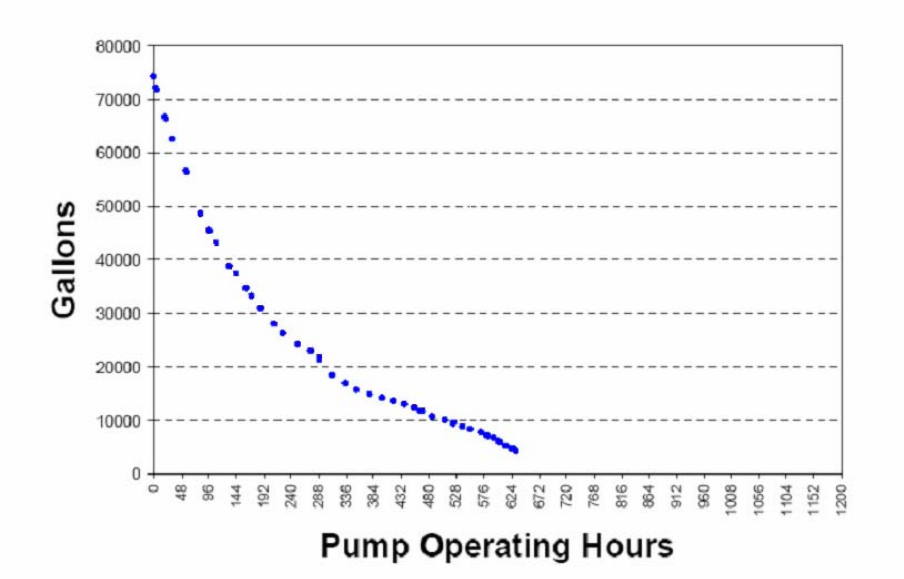


Figure 12-3: Actual Liquid Removal Progress for Tank 41 from March through June 2003 (Flach 2004)

Parameters Important to Optimizing DDA Treatment of Predominantly Insoluble Radionuclides

Insoluble Solids Entrainment During Dissolution

Sr-90, Pu, Np-237, and Am-241 are predominantly found in insoluble solids (sludge). Minimizing the amount of these radionuclides processed through DDA involves minimizing the amount of entrained sludge in the Saltstone Facility feed. Minimizing the amount of entrained sludge in the Saltstone feed is accomplished by:

1. Selecting tanks with less than 3,000 gallons of low-heat waste (LHW) sludge[8] for DDA processing

2. Allowing sludge entrained in the dissolved salt solution to settle below the elevation of the pump intake following salt dissolution to minimize the amount transferred to Saltstone.

Sludge entrainment during dissolution depends on the particle size of the solids, the extent of particle agglomeration or adhesion between particles, the distribution of particle sizes, the distribution of particles within the waste tank, the location of the pump suction relative to the solid particles, the density and viscosity of the liquid phase, and the velocity pattern of the liquid phase during pumping. Phenomena of

[8] High-Heat Waste (HHW) sludges originating from the first canyon cycle have fission product concentrations three orders of magnitude higher than LHW sludges from the second canyon cycle.

entrainment can be described analytically, but values of several parameters are not known, such as, the distribution of solids within the saltcake or the liquid phase during dissolution. Additionally, the distribution of solids within the tank and in the saltcake changes as dissolution progresses. However, experience with salt dissolution at SRS and recent experience at the Hanford Site provides an indication of how the solids will behave during saltcake dissolution. In addition, analysis performed for entrainment of sludge during liquid waste transfers provides an indication of when solids will be entrained during waste transfer. Sample data from dissolved salt solutions and some solid saltcake samples provide an indication of how much insoluble/low solubility solids can be entrained.

Insoluble Solids Behavior in Saltcake Tanks

A saltcake dissolution method similar to DDA was employed to dissolve Tank 20 saltcake. In this case, the saltcake was not deliquified. Dissolution water was added to and removed from the tank in batches as described in the DDA process. Insoluble and low solubility solids remained as the saltcake was dissolved. The solids settled on top of the saltcake. This layer of solids became progressively thicker until successive dissolution water additions became relatively ineffective at dissolving additional saltcake. Approximately two-thirds of the saltcake was dissolved and the remainder was removed several years later after slurry pumps were installed (West 1982). The slurry pumps provided agitation that displaced the solids from the saltcake surface, which resulted in exposing the saltcake to the dissolution water.

Personnel at the Hanford Site recently completed their first saltcake dissolution and removal from Tank S-112 (Barton 2005). This saltcake was deliquified many years before dissolution. In this case, the total liquid inventory in the tank was limited such that the saltcake was not submerged in liquid until most of the saltcake was removed. The water was added in batches and cascaded through the deliquified saltcake. After a short waiting period, the dissolved salt solution was pumped out via saltwell pumping. The wait period progressively increased from 1 to 5 days as dissolution progressed. Photographs/videos from the last 5% or so of the saltcake dissolution show a fine particulate material covering the saltcake. The specific compounds of this apparent low solubility or insoluble material have not yet been identified, but the observations indicate similar behavior of the insoluble solids observed during dissolution of Tank 20.

From these experiences, one can infer that low solubility/insoluble materials, i.e., sludge solids, would tend to settle on top of the saltcake during the DDA process. In addition, the solids layer would become progressively thicker as the saltcake is dissolved, thus, increasing the possibility of entraining more solids during pumping. However, the pump rates remain relatively low which minimizes the liquid phase velocity and, thus, minimizes entrainment. Analysis of the flow pattern around the pump suction for the evaporator systems shows an effective range for entraining sludge solids of about 12 inches from the suction (SRNL 1997).

Based upon these experiences and the plans to dissolve only a portion of the Tank 41, 25 and 28 material during Interim Processing (Mahoney and d'Entremont 2004), it can reasonably be expected that a significant fraction of the low solubility/insoluble materials currently in these tanks will not be removed by the DDA process and will remain in the portion of these tanks that will be processed following Interim Processing. While this solids removal characteristic of the DDA processes is not credited when determining the decontamination achieved by DDA due to uncertainties associated with the insoluble solids content, it does provide further decontamination of the salt waste stored in the tanks. Note as well, that the settling step for dissolved salt solution is included as an integral element of the DDA process to remove the insoluble solids that are not left behind in the tanks associated with Interim Processing.

Actual Saltcake and Dissolved Salt Solution Sample Data

Past samples of saltcake and dissolved salt solution show that insoluble solids content can vary widely as shown in Table 12-2. The solid salt samples indicate the total insoluble solids that might transfer with dissolved salt solution without settling or any other solid liquid separation. The dissolved salt samples show that most of the insoluble solids in saltcake are not likely to transfer with dissolved salt solution or will settle out before transferring. Since these results show total solids, sludge solids cannot be separately identified from the available data.

Table 12-2: Insoluble Solids in Saltcake and Dissolved Salt Solution

Tank	Insoluble Solids Concentration in Salt Sample (mg/L)	Approximate insoluble Solids in Equivalent Dissolved Salt Solution (mg/L)
Tank 38 saltcake (Drumm and Hopkins 2003)	13,700	3,900
Tank 41 saltcake (Drumm and Hopkins 2003)	13,000	4,580
Tank 37 saltcake (Drumm and Hopkins 2003)	25,800	8,720
Tank 24 dissolved salt solution (Fowler 1982)	See Note 1	27,300
Tank 24 dissolved salt solution (Walker and Hamm 1983)	See Note 1	103
Tank 1 saltcake (Fowler 1981a)	19,800	6,600
Tank 20 dissolved salt solution (Fowler 1981c)	See Note 1	none detected
Tank 19 saltcake (Fowler 1980)	51,000	17,000
Tank 19 dissolved salt solution after Transfer to Tank 18 (Fowler 1981b)	See Note 1	< 100

Note 1: Analysis performed on dissolved salt solution sample, therefore no value for saltcake

From experiences identified and the available sample data for similar conditions, minimal solids are expected to be entrained in the dissolved salt solution. However, the relative amount of insoluble solids in the saltcake show that unexpectedly high entrainment of insoluble solids is possible, thus, requiring a settling step after dissolution until enough dissolution experience shows this step to be unnecessary.

Insoluble Solids Settling

Settling rate of the entrained solids is dependent on particle size, particle density, particle density in the liquid phase, liquid density, liquid viscosity, and time. The liquid phase properties are reasonably known or predictable, but the solid phase properties are unknown primarily because measurements of this type have not been made on dissolved salt solution because, historically there has been very limited dissolution of salt. However, solid phase property measurements exist for sludge solids. Since the sludge solids contain the majority of the fission products and actinides (d'Entremont and Drumm 2005), other solids settling are not as important to the radionuclide removal efficiency.

Effectively, settling in a waste tank can be described in terms of the downward movement of an interface with time. The liquid above the interface is clear of any

solids larger than a certain minimum size. The minimum size is picked such that more than 99% of the sludge particles are larger than the minimum. The liquid above this interface is effectively decontaminated of sludge particles and only the soluble radionuclides remain. The rate of change in the interface level was estimated for the dissolved salt solution from the first dissolution tank. The rate is expected to be similar for subsequent tanks, but a detailed estimate will be made on a tank-by-tank basis before dissolution occurs. The first tank settling rate is estimated in Table 12-3 (Gillam 2005):

Table 12-3: Sludge Solids Settling Rates

Fraction of Solids Removed	Settling Rate, in./day
0	37.00
0.500	16.00
0.667	9.00
0.750	6.84
0.800	5.54
0.900	2.95
0.935	2.03
0.950	1.65
0.964	1.28
0.975	1.00
1	0.35

For example, in order to remove 0.667 or 66.7% of the entrained solids from a dissolved salt solution batch 300 inches deep (assuming the settled sludge layer is less than 6 inches deep and approximately 2 feet below pump suction if the pump suction is at 30 inches above the tank bottom), the solids must be allowed to settle 33 days at a settling rate of 9 inches per day. The actual settling time is adjusted to allow adequate time to settle solids to meet SPF process requirements and balance the need to create enough working volume in the tank farm to maintain waste process operations. The current baseline case is a 30-day settling period.

Selection of DDA Tanks

Tanks were selected to undergo the DDA process during Interim Processing using the following criteria:

1. Tanks selected for DDA should be Type III tanks. Type III tanks meet current Environmental Protection Agency (EPA) requirements for full secondary containment and leak detection and are therefore the only tanks deemed suitable for additional waste receipt. Selecting Type III tanks for DDA frees compliant tank space required to receive additional waste streams created during SWPF batch

preparation, waste removal and sludge batch preparation, and H-Canyon legacy material stabilization operations.

2. Tanks selected for DDA should not be used for an operational function vital to Tank Farm processes such as evaporator systems or sludge batch preparation. Selecting a tank serving such a function for DDA would incur cost, schedule, and system impacts in order to set up another tank to replace its operational function and, in most cases, would not be physically or chemically possible.
3. Tanks selected for DDA should not be high-heat waste (HHW) tanks. HHW sludges originating from the first-canyon cycle have fission product concentrations three orders of magnitude higher than low heat waste (LHW) sludges from the second-canyon cycle. In order to minimize the amount of these radionuclides carried over through the DDA process, HHW tanks were not selected for DDA.
4. Tanks selected for DDA should contain minimal amounts of insoluble solids. Insoluble Tank Farm solids (sludges) contain larger amounts of strontium and actinides than the supernate phases. In order to minimize the amount of these radionuclides sent to SDF, only tanks estimated to contain minimal amounts of sludge (<3K gal.) were selected for DDA.
5. Tanks selected for DDA should have lower activity supernate waste. In order to identify the tanks with lower activity supernate waste, it is important to identify tanks that are relatively low in Cs-137. Cs-137 is highly soluble and constitutes the bulk of the curies that will be sent to the SDF.

Also, a correlation exists between the Cs-137 concentration and the concentration of other soluble radionuclides important to SDF performance such as I-129, Tc-99, and Sr-90 (Hill 2005, Tran 2005, and Hester 2004).

Using these criteria, seven tanks were identified with Tanks 41, 25, and 28 chosen as the tanks most suitable for DDA processing. The following table lists the Type III tanks in ascending order of their supernate Cs-137 concentration. Table 12-4 designates which tanks contain sludge volumes greater than 3,000 gallons, the type of sludge (HHW or LHW) in each tank, and any current operational function of the tanks in addition to waste storage.

Table 12-4: Selection of Tanks for DDA Processing

Tank	Current Operational Function	Sludge Type	>3k gallons sludge?	Cs-137 Supernate Concentration (Ci/gal)	Suitable for DDA consideration?
50	Saltstone Blending / Feed Tank	n/a		4.9E-05	No
40	DWPF Sludge Prep / Feed Tank	HHW	Yes	5.8E-02	No
48	Precipitate Storage	n/a		6.1E-02	Yes
51	DWPF Sludge Prep / Feed Tank	HHW	Yes	1.8E-01	No
33		HHW	Yes	3.0E-01	No
41		LHW		3.8E-01	Yes
43	2H Evaporator System Feed / Vent Tank	HHW	Yes	6.7E-01	No
38	2H Evaporator System Drop Tank	n/a		7.8E-01	No
42		HHW	Yes	9.7E-01	No
47	2F Evaporator System Vent Tank	LHW		1.6E+00	No
29	3H Evaporator System Vent Tank	n/a		3.3E+00	No
25		n/a		3.5E+00	Yes
39	H-Canyon Receipt Tank	HHW	Yes	3.6E+00	No
34		HHW	Yes	4.0E+00	No
27	2F Evaporator System Drop Tank	LHW		4.1E+00	No
28		n/a		4.5E+00	Yes
26	2F Evaporator System Feed Tank	LHW	Yes	4.5E+00	No
45		n/a		5.0E+00	Yes
44		n/a		5.3E+00	Yes
46		n/a		7.5E+00	Yes
35		HHW	Yes	7.7E+00	No
32	3H Evaporator System Feed Tank	HHW	Yes	9.1E+00	No
31		n/a		1.1E+01	Yes
37	3H Evaporator System Drop Tank	n/a		1.3E+01	No
49	SWPF Feed Tank	n/a		1.3E+01	No
30	3H Evaporator Alternate Drop Tank	HHW		1.7E+01	No
36		HHW		2.1E+01	No

Note: Data taken from November 2003 Waste Characterization System (WCS). It was assumed that the Cs-137 concentration is in equilibrium throughout each of the tanks.
Sludge Type "n/a" indicates no appreciable sludge present.

Although Tank 48 will require all the steps associated with DDA processing, disposing of the unique waste in Tank 48 is critical during Interim Processing. As discussed on page 16 of the Salt WD, the Tank 48 waste consists of approximately 0.24 Mgal of a relatively low-activity salt solution containing potassium and cesium tetraphenylborate (TPB) salts generated during an earlier unsuccessful effort to prepare salt waste for disposal, known as the In-Tank Precipitation (ITP) process. The organic nature of TPB salts requires them to be stored separately from other tank waste. This is because TPB can break down into benzene and other organic compounds, and can form a potentially explosive mixture in the vapor space of a waste tank if not carefully managed. Other tanks are not equipped with safety systems required to manage this flammable mixture. Accordingly, all of the space in the 1.3 Mgal Tank 48 is being entirely used to store the 0.24 Mgal of TPB salts.

In addition, this waste cannot be processed through DWPF because the breakdown of TPB in sufficient quantities in the DWPF melter could pose safety concerns. Currently, there is no practically available or contemplated technology that could be used to remove additional radioactivity and dispose of that radioactivity using DWPF. Technologies that were considered for the treatment of the waste in Tank 48 are discussed in the response to NRC Comment 13. Accordingly, the waste in Tank 48 (see pages 40-42 of the Salt WD for more information on the Tank 48 waste) will be processed without further removal of radionuclides by aggregating the Tank 48 stream with another salt waste stream, currently planned to be the low-activity liquid recycle waste stream from DWPF. The two waste streams will be aggregated to ensure the processing limits for allowable organic content at SPF are not exceeded. These limits are contained in the waste acceptance criteria for the Saltstone Processing Facility. This is further discussed in the response to NRC Comment 37. The aggregated low-activity waste stream will then be transferred to the Saltstone Facility feed tank. Dispositioning the waste in Tank 48 during Interim Salt Processing is critical because:

- Dispositioning the waste in Tank 48 allows the use of up to 1.3 Mgal of space in this tank. Without this space, there is not enough space in Type III tanks to stage dissolved salt SWPF feed batches.

- The location of Tank 48 makes it an integral part of staging feed for SWPF.

- Tank 48 is the planned feed tank for the Actinide Removal Process (ARP) and the Modular Caustic Side Solvent Extraction Unit (MCU) process.

As the table demonstrates, Tanks 41, 25, and 28 are the lower activity LHW Type III tanks that have minimal amounts of sludge and no operational function that precludes them from being ideal candidates for DDA. Thus, Tanks 41, 25, and 28 were selected as the tanks containing some of the lowest activity waste most suitable for initial DDA processing. Selecting these lower activity waste tanks for DDA will minimize the amount of radionuclides sent to SDF in the DDA waste.

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NRC Comment 15: The basis for the amount of sludge entrained in waste processed through the DDA process is unclear. The uncertainty in the concentration of key radionuclides, particularly for the DDA waste stream, is not provided and the point estimates are not clearly reasonably conservative.

Basis: On page 3-8 of Reference 3 it is noted that the waste concentrations for Low Curie Salt are based on the assumption that 300 mg/L of sludge is entrained in salt solutions derived from salt processing. In Reference 6 it is noted that the salt waste in Tank 41H would contain more than 400 mg/L of entrained sludge. Concentrations of some radionuclides that strongly influence the results, including Sn-126, will be sensitive to the amount of sludge entrained. Page 44 of Reference 4 lists the concentration of TRU radionuclides in DDA waste as 64% of the limit, but it is unclear from the information provided what key assumptions may have been made in the derivation of this value. For the overall salt waste treatment process, uncertainty of 3 to 5 MCi is estimated for the total inventory (essentially all Cs-137), but uncertainty is not provided for other highly radioactive radionuclides that drive the risk.

Path Forward: Provide the basis for the amount of sludge (and its associated radiological composition) that will be entrained in salt solutions sent to saltstone. Provide the uncertainty in the inventory of highly radioactive radionuclides (e.g., Sn-126, Tc-99, Np-237, I-129, Se-79) in saltstone, considering uncertainty in: 1) settling removal efficiencies, 2) sludge entrainment during salt processing, 3) sludge radiological compositions, and 4) saltcake concentrations. The response should clearly indicate whether the information is from direct observation (therefore less uncertain) or indirect methods (therefore more uncertain). Provide a summary of the direct measurement data of the radiological composition of saltcake.

SRS Response: The characterization information, used in the draft Salt Waste Disposal Waste Determination (WD) [NRC Reference 4], provided input to the salt disposal planning process. Saltstone Disposal Facility (SDF) Waste Acceptance Criteria (WAC) have been developed to protect the nuclide inventory limits established in the 2005 Special Analysis. Compliance with these Waste Acceptance Criteria is confirmed through sample and analysis of the material that is being sent to SDF. Therefore, although uncertainty exists in characterization of the waste and in assumptions for predicting attributes such as processing efficiencies and settling rates, these uncertainties are rendered inconsequential by actual sample analysis of the waste.

The assumed entrainment rate in d'Entremont and Drumm (2005) for insoluble solids in the salt solution is 600 mg/L as it arrives in the Deliquification, Dissolution, and Adjustment (DDA) settling tank (see pages

12 – 16 of the draft Salt Waste Disposal Waste Determination for a description of the DDA process). The concentration of insoluble solids in the salt solution source tank can be considerably higher (e.g., the Tank 41 sample referred to in the Basis of this Comment).

The concentration of 600 mg/L is the entrainment rate that has historically been used as a design bases for salt planning. Although this concentration was originally used as a processing upper bound to protect filter performance in the now defunct In-Tank Precipitation process, the limited available characterization information on insoluble solids carryover in dissolved salt solution described below supports that this is not an unreasonable assumption. Two samples of dissolved salt solution have been obtained from tanks that received dissolved salt solution. The results are shown in Table 15-1. These samples are equivalent to salt solution samples obtained from the settling tanks described in the WD. In both of the examples shown below, bulk saltcake was dissolved using large mixing devices. The saltcake in these tanks formed in a similar manner as the salt in the DDA source tanks (i.e., precipitation of salts due to cooling of concentrated supernate). Therefore, the sludge entrainment in the bulk salt in Tanks 24 and 19 should be similar to the sludge entrainment in the DDA source tanks. Since large mixing devices were used to aid in salt dissolution, the insoluble solids carried over into the tank receiving the dissolved salt solution is expected to be higher than the insoluble solids carried forward in the DDA process since the insoluble solids are allowed to settle instead of being suspended by the mixing device.

Table 15-1. Settling Tank Solids Concentrations

Source Tank	Settling Tank Dissolved Salt Solution Insoluble Solids (mg/L)
Tank 24 dissolved salt solution (Walker and Hamm 1983)	103
Tank 19 dissolved salt solution (Fowler 1981)	< 100

Dissolved salt solution has recently been transferred from Tank 41 (DDA source tank) to Tank 49 (dissolved salt solution settling tank) and two samples were obtained from Tank 49 prior to the 30-day settling period. One sample was obtained at an elevation of 20” from the tank bottom; the other was obtained from an elevation of 95” from the tank bottom. The primary purpose of these samples was to determine the settling rate of solids that were carried over. Figure 15-1 shows the samples as received at the Savannah River National Laboratory for analysis (Martino 2005). As can be seen in Figure 15-1, the sample liquid was very clear and lacked visible insoluble solids. Insufficient solids were present to perform settling studies but the insoluble solids concentration prior to settling was clearly less than 600 mg/L. With an

insoluble solids concentration of 600 mg/L, solids would be visible in the sample.

Figure 15-1. Tank 41 Dissolved Salt Solution Obtained from Tank 49 prior to Settling.

Although the data are limited, it can be seen from the information in Table 15-1 and Figure 15-1 that 600 mg/L is not an unreasonable assumption. Also, it is important to keep in mind that the validity of this assumption will be confirmed by sampling of the dissolved salt solution in the settling tank for each batch prior to transferring this material to the Saltstone Feed Tank.

The response to NRC Comment 11 discusses highly radioactive radionuclides. The initial (pre-settling) entrained solids concentration in the settling tank is only important during the DDA portion of salt processing since the entrained solids will be filtered during the Actinide Removal Process/Modular CSSX Unit (ARP/MCU) and Salt Waste Processing Facility (SWPF) processes. The relative efficiencies for radionuclide removal are discussed in the response to NRC Comment 11.

For Cs-137, approximately 300 curies of the 1.2 million curies sent to SDF during the DDA process are due to entrained solids and therefore the uncertainty in the entrained solids concentration is inconsequential (d'Entremont and Drumm 2005). Cesium-137 in solution is well characterized through direct observation (greater than 1000 sample analyses).

For alpha-emitting TRU, approximately 280 curies of entrained solids are projected to be sent to the SDF during the DDA process (d'Entremont and Drumm 2005). This is ~2% of the nearly 15,000 total curies sent to the SDF during the entire salt processing program (d'Entremont and Drumm 2005).

Therefore, the settling tank solids concentration has a minimal impact on total alpha-emitting TRU curies in the SDF. Of the ~280 alpha-emitting TRU curies sent to SDF from entrained solids (during DDA), ~270 curies are from Pu-238, Pu-239, and Pu-240. These plutonium concentrations in sludge are characterized by direct observation (sample analysis performed in the generating facilities, F & H Canyons, prior to transfer of the waste to the waste tanks). The remaining curies (primarily Am-241) are characterized indirectly through a combination of sampling and process history.

The removal efficiencies for soluble Sr-90 are much greater than the removal efficiencies for soluble alpha-emitting TRU in the ARP/MCU and SWPF processes and, therefore, the fraction of Sr-90 inventory in SDF due to insoluble solids is greater than the fraction for alpha-emitting TRU. Approximately 5100 curies of the 7500 curies of Sr-90 sent to the SDF during the entire salt processing program are due to entrained solids and therefore any uncertainty in the entrained solids concentration significantly contributes to the total Sr-90 curies in the SDF. If the insoluble solids concentration assumption was low by a factor of 2, the total Sr-90 inventory could be as high as 12,600 curies. This increase is inconsequential since the Vault 4 disposal limit for Sr-90 is 2.4E+16 curies (Cook et al. 2005). Sr-90 in the sludge is well characterized through direct observation (sample analysis performed in the generating facilities, F- and H-Canyons, prior to transfer of the waste to the waste tanks). Once again it is important to remember that these predictions will be confirmed through sample analysis of the dissolved salt solution in the settling tank.

For the other radionuclides considered in the draft WD (i.e., Se-79, Tc-99, Sn-126, I-129, Np-237), the fraction of predicted SDF curies resulting from entrained solids carryover during DDA is displayed in Table 15-2. The concentrations of these constituents (with the exception of Np-237) in the sludge are determined indirectly and are based on fission product yield information (Georgeton and Hester 1995). The Np-237 concentration in sludge is determined by direct observation (sample analysis performed in the

generating facilities, F- and H-Canyons, prior to transfer of the waste to the waste tanks). It can be seen that the contribution to the total SDF inventory from entrained solids carryover is insignificant and therefore, the uncertainty in entrained solids concentrations in the salt solution receipt tank has a negligible effect on the inventories of these nuclides in SDF.

Table 15-2. Percentage of SDF Curies due to DDA Entrained Solids (d'Entremont and Drumm 2005)

Nuclide	Entrained Solids from DDA	Total Predicted Curies to the Saltstone Disposal Facility	Percentage due to DDA Entrained Solids
Se-79	7.0E-02	8.9E+01	0.1%
Tc-99	1.3E+00	3.3E+04	0.004%
Sn-126	9.0E-02	4.5E+02	0.02%
I-129	5.0E-06	1.8E+01	<0.001%
Np-237	1.0E-02	2.1E+00	0.5%

Table 15-3 below provides the requested summary of direct measurement data of the radiological composition of saltcake. These sample results represent the equivalent of bulk saltcake after the Deliquification portion of the DDA process. The constituent concentrations listed in Table 15-3 should not be construed to represent constituent concentrations in the feed to the SDF. Significant settling will occur in the DDA source tank and settling will occur in the dissolved salt solution settling tank. As a result, concentrations of insoluble radionuclides sent to the SDF will be significantly lower than the concentrations found in dry saltcake.

Table 15-3. Major Radionuclide Concentrations in Dry Saltcake (Drumm et al. 2004)

pCi/ml	Tank 2	Tank 3	Tank 10	Tank 29	Tank 38	Tank 41
Tc-99	1.1E+04	6.0E+04	5.4E+04	6.8E+04	8.4E+03	5.0E+03
Cs-135	5.9E+02	8.8E+02	1.3E+02	4.6E+02	5.1E+01	2.4E+02
Th-230	1.3E+02	1.3E+03	6.2E+03	6.5E+02	1.8E+03	1.1E+03
Th-232	5.6E-02	1.1E-02	4.5E+00	1.3E-01	4.8E-01	5.1E-03
U-233	6.4E+01	6.3E+02	2.0E+04	3.8E+03	8.6E+02	1.5E+02
U-234	4.2E+01	4.0E+02	4.1E+03	5.0E+03	1.2E+04	NVR
U-235	5.0E-01	1.5E-01	2.9E+01	1.4E+01	1.4E+01	NVR
U-236	3.0E+00	4.2E+00	2.7E+02	1.9E+02	6.2E+01	NVR
Np-237	4.7E+00	4.6E+01	1.1E+03	1.4E+02	6.5E+02	2.3E+01
U-238	1.4E+01	6.7E+00	6.2E+01	9.8E+00	3.0E+02	NVR
Pu-239	1.0E+04	4.0E+03	2.7E+04	1.9E+03	6.1E+04	3.4E+03
Pu-240	1.1E+04	1.5E+04	7.0E+04	7.3E+03	5.0E+04	1.3E+04
Pu-241	8.5E+05	6.7E+06	3.2E+07	3.3E+06	9.2E+06	5.7E+06
Pu-242	2.5E+01	2.5E+02	1.2E+03	1.2E+02	1.2E+03	2.1E+02
C-14	3.6E+03	8.2E+03	4.8E+04	2.1E+03	1.1E+04	4.7E+02
Sr-90	2.6E+06	1.6E+06	1.2E+08	2.8E+07	1.0E+07	NVR
Cs-137	NVR	3.9E+07	3.1E+06	NVR	NVR	1.6E+07
Pu-238	2.2E+04	6.2E+03	2.8E+06	2.4E+06	7.0E+06	2.7E+04
Pu-239/240	1.3E+04	4.1E+03	3.8E+04	3.6E+04	1.3E+05	1.8E+03
Am-241	7.3E+03	2.7E+03	5.1E+04	2.3E+04	NVR	NVR

NVR – No Value Reported in sample analysis report

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NRC
Comment 16: It is not clear why the concentrations of some of the most risk significant highly radioactive radionuclides, as reported in current waste inventory projections, are significantly lower than the concentrations reported in earlier projections even though the overall radiological composition increased substantially.

Basis: Comparison of the nominal blend of waste in 1992 (pg. 2-66 of [1]) with the Low Curie Salt (LCS) solution in 2002 (pg. 3-8 of [3]) shows that the concentrations of most of the radionuclides in the LCS waste were expected to be significantly higher than they were in the nominal blend in 1992. However, the inventories for Tc-99, Se-79, I-129, and C-14 all decreased. In addition, the concentration of Sn-126 increased by a smaller amount than would be expected based on the increases in the inventories of other radionuclides. Tc-99, Se-79, I-129, C-14, and Sn-126 are most of the more risk significant radionuclides. Reference 7 indicates that the Tc-99 concentration projected for saltstone was 36 times larger than projected in Reference 5.

Path Forward: Provide an explanation for the evolution of the inventory of key radionuclides over time. Explain why the concentration of the radionuclides given above decreased substantially or did not increase in proportion to most of the radionuclides in the more recently estimated saltstone compositions [3, 5] as compared to the composition estimated in 1992.

SRS Response: DOE's understanding and degree of characterization have evolved over time. The early stages of characterization focused on minimizing the discard of and fully accounting for special nuclear materials like thorium, uranium, neptunium, and plutonium, and on the need for safe storage of waste. DOE's methodologies and standards for assuring nuclear safety have evolved and characterization needs have evolved with them.

In the 1992 PA, the preparation of a PA for waste disposal actions planned at the SDF drove the development of characterization for some additional radionuclides that were important in the performance assessment process. These characterizations were based upon the best available data at that time and were derived from both sample data and process histories. For example, the algorithm for C-14 concentration for all soluble waste was estimated based on a single Tank 8 supernate sample (Fowler 1982, page 3). Likewise, the I-129 inventory of the SRS waste tanks was estimated at 9.1 curies (Fowler and Cook 1984) based on two supernate samples, two salt samples, and canyon processing records. In a similar manner, 24 sample analyses were used to calculate a total soluble Tc-99 inventory of approximately 2E+4 curies (Fowler et al. 1984).

In 1995, the Waste Characterization System (WCS) was developed to track radioactive and non-radioactive constituents in the sludge, supernate, and salt phases.

The characterization information was based on a combination of processing records, theoretical fission yield relationships, and sample data for accountable materials (i.e., thorium, uranium, neptunium, and plutonium). At the time, the WCS was developed to assist in characterizing the waste in order to ensure safe storage. The radionuclide content of the sludge has the largest impact on inhalation dose potential, hydrogen generation rate, and hydrogen retention in the waste tanks. As such, significant effort was placed on characterizing the sludge phase of the waste. The effort to characterize the soluble portion of the waste (supernate and saltcake) was focused on Cs-137/Ba-137m and Pu-238 since these were the most significant contributors in supernate to safe storage considerations. Cesium-137 and its daughter Ba-137m are important because they are the source of 99 % of the radioactivity in the supernate and are the largest external dose contributors. Plutonium-238 is important because it is the major contributor to the inhalation dose of supernate. The soluble portions of most radionuclides (e.g., Se-79, Tc-99, Sn-126, and I-129) were not tracked in WCS because they were not significant with respect to safe storage.

Both the 2002 Special Analysis (SA) (Cook et al. 2002) and the 2005 Dose Assessment (Cook 2005) used data extracted from WCS to predict the SDF inventory. (Note: The 2005 Dose Assessment used data extracted from WCS in May 2004.) Due to the limitations in soluble nuclide characterizations described above, these projections were lower than the predictions in the 1992 PA. Some of the constituents (e.g., Se-79, Sn-126) were only predicted based on their presence in the entrained sludge. For others (e.g., Tc-99, I-129) the soluble inventories were only calculated for the limited number of tanks for which sample results were available.

In the fall of 2004, an initiative was begun to expand WCS to upgrade it to more accurately represent radionuclides important to disposal and closure actions that had not been important to safe storage considerations. A review of available salt sample data determined that C-14 was present in the drained salt samples and, for characterization purposes, a constant saltcake C-14 concentration of 6.04E+3 pCi/ml was recommended (Drumm et al. 2004). Also in the fall of 2004, a set of supernate concentration algorithms was developed to support the Salt Waste Processing Facility design effort. These algorithms were based on seven recent supernate sample analyses and were developed for H-3, C-14, Co-60, Ni-63, Sr-90/Y-90, Tc-99, and I-129 (Hester 2004). Since these algorithms were developed to support design activities they were intentionally conservative in projecting concentrations. For a few

of the radionuclides (e.g., Tc-99, I-129), the algorithms were too conservative to be used to reasonably predict SDF inventories. For others (e.g., C-14), although still conservative (C-14 = 1.76 E+3 pCi/ml), the algorithms seemed reasonable for use in predicting the SDF inventories. Unfortunately, neither the saltcake nor the supernate concentration algorithms were available when the WCS data was extracted to support development of the 2005 Dose Assessment. Using these algorithms, the soluble C-14 inventory sent to the SDF is predicted to be approximately 5.2 E+2 curies.

Concurrent with the development of the 2005 Dose Assessment, the I-129 supernate inventory was characterized by using sample data to determine the average supernate Cs-137 concentration to I-129 concentration ratio (Tran 2005). Since Cs-137 is well characterized in the supernate, this method allows the I-129 supernate inventory to be estimated for each tank that does not have sample data available. Utilizing this method, a total supernate inventory of 1.7E+1 curies was established. This value was included in the 2005 Dose Assessment.

Further work in 2005 provided methods for determining the total estimated soluble inventories for Se-79, Tc-99, and Sn-126 (Hill 2005). These inventories were developed using a combination of process knowledge and available sample data. The soluble inventories for these radionuclides predicted to be sent to the SDF are approximately 8.9E+1, 3.3E+4, and 4.5E+2 curies, respectively. These inventories (along with the revised C-14 soluble inventory) have been used as input to a revised dose assessment. A description of the dose assessment and the associated key assumptions is contained in the Performance Objective Demonstration Document (PODD) (Rosenberger et al. 2005). As with the 2005 Dose Assessment, the contribution from entrained sludge has been added to the soluble inventories.

Recently, increased effort has been placed on characterizing the soluble portions of radionuclides that were in the past consider to be insignificant due to their negligible impact on safe storage of the waste. These characterizations much more closely estimate soluble inventories than previous efforts, especially the characterizations that were used as input to the 2002 SA (Cook 2002) and the 2005 Dose Assessment (Cook 2005). Lastly, it is important to remember that these inventories are used to project saltstone inventories for planning purposes. The radionuclide content of the material sent to the Saltstone Processing Facility will be confirmed through sample and analysis.

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NRC Comment 37 The basis for performance of saltstone containing Tank 48 waste (TPB organics) is not provided. It is not clear what the basis is for the limit on allowable organic content in the Waste Acceptance Criteria (WAC) for the Saltstone Processing Facility (SPF).

Basis: Reference 4 (pg. 16) indicates that Tank 48 waste will be sent directly to saltstone without treatment, but that the waste from Tank 48 will be mixed with other streams of low activity waste so that the processing limits for allowable organic content at the SPF are not exceeded. The physical characteristics of saltstone and its durability with respect to the retention of radionuclides may be significantly different when produced with the organic material from Tank 48 waste. For example, biodegradation of an organic-containing wasteform could represent a degradation mechanism that has not been evaluated in the testing to date.

Path Forward: Provide the basis for the performance of the saltstone (including the physical properties) and provide the basis for the limit on allowable organic content in the WAC for SPF.

SRS Response: The disposition of Tank 48 salt waste with its associated organic material (mainly potassium and cesium tetraphenylborate) at the Saltstone Disposal Facility (SDF) is a forecasted activity; disposal of similar material has not historically occurred nor been anticipated in the SDF. For this reason, the laboratory studies and associated analyses to support the development of the revised limits associated with the Documented Safety Analysis (DSA) and the Waste Acceptance Criteria (WAC) for both the Saltstone Processing Facility (SPF) and the SDF are not complete. The response to NRC Comment 6 describes this process for establishing DSA limits in more detail. Such changes to both the DSA and WAC limits will not occur without a rigorous, disciplined process to address both the short-term and long-term impacts to the health and safety of occupational workers, the public and the environment.

Changing the WAC for Saltstone to allow it to receive Tank 48 waste follows the thorough process outlined in WSRC Manual S4 Procedure ENG.08 Rev. 2 (WSRC 2005). This manual requires that a proposed change to the WAC to permit acceptance of a new waste stream go through a formal review and approval process. A primary element of that process is a system impact analysis which is performed to characterize the consequences that the proposed change has on the downstream facilities. The system impact analysis is a key component of the WAC and Waste Compliance Plan (WCP) revision process as it identifies the potential for downstream impacts of introducing a new waste stream and initiates the formal engineering review process and the resulting technical evaluation of impacts of the proposed change.

Using the process outlined in ENG.08 to assess the impacts of the forecasted Tank 48 waste stream, there are four primary criteria that form the basis for establishing organic limits in the Saltstone WAC:

1. The salt solution and the resulting grout must be able to be safely processed in the SPF and disposed of in SDF;
2. The final saltstone grout must be characteristically non-hazardous in accordance with RCRA regulations;
3. The respective concentrations of the organic material present (mainly potassium and cesium tetraphenylborate) must not adversely impact the facilities capability to demonstrate compliance with the performance objectives in 10 CFR 61, Subpart C;
4. The organic concentrations must not adversely impact processibility of the salt solution and/or grout.

Safety:

The currently approved WAC for the SPF (Chandler 2004) lists two acceptance limits associated with Tank 48 organics. Both of these limits are provided to protect assumptions made in the Time-to-LFL (Lower Flammability Limit) calculation for the Salt Feed Tank (SFT) used in the SPF Documented Safety Analysis (DSA) (WSRC 2004). The first limit is a maximum concentration limit set on tetraphenylborate (TPB) (both soluble and insoluble) at 30 mg/L. The second is a limit on the maximum allowed benzene generation rate from the decomposition of TPB at 0.092 mg/L/hr. It is anticipated at this time that both of these limits can be increased. The actual limits will be determined following the laboratory testing and analyses.

As is described in Section 7.2.3.14 of the Draft Waste Determination Document (WD) (DOE 2005), before the SPF process is modified including the addition of a new waste stream or modification to the existing salt waste stream, a Consolidated Hazards Analysis (CHA) is performed to identify potential hazards associated with the modification, classify those hazards and evaluate the consequence and frequency of each of the hazards identified. The DSA will document the analysis of hazards identified through the CHA process and will provide the basis for any controls required to achieve safe operations in the SPF. Those controls will be documented in the Technical Safety Requirements (TSR) document for the SPF. If the rigorous analysis process determines that the WAC limits can safely be changed based on the previously described four criteria, then and only then will the WAC limits be revised. The viability of the disposal of Tank 48 salt waste in the SDF will then be dependent on the ability to demonstrate that salt solution from Tank 48 can meet these WAC limits.

Laboratory testing and analyses are on-going to support the evaluation above. Effects of tetraphenylborate decomposition as a function of Saltstone curing temperature and time

were investigated using a nonradioactive Tank 48 surrogate composition. Potassium tetraphenylborate was found to decompose in the Saltstone matrix in samples cured above 75º (Cozzi et al. 2005). Decomposition products include benzene (Cozzi and Zamecnik 2004). An extended testing program of Saltstone made with actual Tank 48 material and simulant is being conducted to evaluate the decomposition of the tetraphenylborate during curing. To date, tetraphenylborate does not appear to affect the durability of Saltstone samples cured below 75ºC. Consequently, one option for operating the facility which is currently being explored is to control the Saltstone curing temperature to a value low enough to prevent decomposition of the potassium tetraphenylborate.

Non-Hazardous Status:

The SDF is permitted as a non-hazardous landfill by the State of South Carolina. Thus, only material deemed non-hazardous in accordance with Resource Conservation and Recovery Act (RCRA) regulations may be placed in the disposal vaults. As described below, testing performed to date indicates that the presence of TPB at concentrations in compliance with the current WAC limits have met the Toxicity Characteristic Leaching Procedure (TCLP) requirements resulting in the saltstone grout being classified as non-hazardous. Any future changes to increase the limits for organics will require verification that the non-hazardous nature of the waste form is not changed by the increase in organic limits.

Testing performed to date includes a preliminary TCLP test. Processing aggregated tank waste containing the actual Tank 48 material was demonstrated and preliminary testing indicated that extraction of mercury in the TCLP leachate is not accelerated by the organics present in the actual waste (Cozzi 2004). (Mercury is the only TCLP metal present in concentrations of concern in the waste.)

Performance Objectives Met:

Literature search to date has not identified any specific studies that have been performed on the long–term effects of organic material on grout performance (i.e., effects of biodegradation). However, the effects of grout degradation over time have been evaluated. In the Special Analysis for Vault 4 (Cook et al. 2005), the long-term effects of grout degradation were evaluated by changing the hydraulic conductivity of the grout as the vaults aged. The sensitivity of grout performance to these changes in hydraulic conductivity is discussed in the response to NRC Comment 19. These sensitivity analyses provided reasonable assurance that the increased hydraulic conductivity associated with hypothetical saltstone grout degradation would not result in exceeding the performance objectives in 10 CFR 61, Subpart C. See the response to NRC Comment 57 for further discussion.

Processibility:

The salt solution has specific physical and chemical properties that must be met in order to process the material through the SPF without causing system upset. These

properties are specified in the SPF WAC and include pH of the aqueous solution, the sodium ion concentration, the temperature of the salt solution, and the total mass of insolubles. In addition, the WAC states that aqueous waste sent to the SPF shall not contain or generate volatile organic materials at concentrations that can produce, at equilibrium, vapors in the flammable or explosive range during normal storage, treatment, or disposal operations in the Saltstone Facility. Any future changes to increase the limits for organics will require verification that the processibility of the waste form is not changed by the increase in organic limits.

Conclusion:

The establishment of the limits as described above will be completed prior to the initiation of the processing of the Tank 48 material. The timing for this processing is based on the Interim Salt Processing Strategy Planning Baseline (Mahoney et al. 2004). Upon determination of the most restrictive criteria (safety, non-hazardous, performance and processibility) for TPB/benzene, a WAC limit will be established that is protective of the most restrictive criteria for organics. The WAC undergoes a formal review and approval process by Operations and Engineering Management prior to implementation (WSRC 2005). Likewise, upon revision of the WAC, the WCP for the sending facility will be revised (as required) to document the means that the sending facility will use to demonstrate compliance with the WAC. The revised WCP will go through a formal review and approval process including review by the receiving facility (Saltstone). Upon successful execution of the process described in the WCP to demonstrate compliance with the WAC, the waste stream can be sent from Tank 50 to the Saltstone Facility.

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NRC
Comment 57: The potential effects of organic chemicals in the Tank 48 waste and in unintentional contamination from the ARP and CSSX treatments on saltstone durability and radionuclide retention in saltstone should be explained.

Basis: Experiments of saltstone durability have been based on samples prepared with simulated saltstone solutions that did not include the organic chemicals present in Tank 48 waste or chemicals that could be unintentionally carried over from ARP or CSSX treatments. Thus the potential effects of these chemicals and their degradation products on saltstone durability should be discussed.

Furthermore, the organic chemicals in Tank 48, as well as the organic chemicals used in the ARP and CSSX process, were designed to react with metals. It is unclear whether tetraphenylborate present in Tank 48 waste, or monosodium titanate and calixarene molecules that could be unintentionally carried over from the ARP and CSSX process could interfere with the precipitation of Tc_2S_7 or result in the formation of radionuclide complexes that would have a higher mobility than the uncomplexed radionuclides. Consequently, the effects of chemicals in the Tank 48 waste and any chemicals unintentionally carried over from the ARP and CSSX processes on the retention of radionuclides in saltstone should be addressed.

Path Forward: Discuss the expected effects of the organics in Tank 48 waste on saltstone durability and radionuclide retention. Provide an estimate of the types and amounts of organic chemicals that are expected to be carried over from the ARP and CSSX treatments into saltstone. Discuss the potential effects of any solvents and extractants carried over from the ARP and CSSX treatments into saltstone on saltstone durability and radionuclides retention.

SRS Response: The disposition of Tank 48 salt waste with its associated organic material (mainly potassium and cesium tetraphenylborate) as well as the disposition of the low-level salt waste streams from Actinide Removal Process (ARP), Modular Caustic Side Solvent Extraction Unit (MCU), and the Salt Waste Processing Facility (SWPF) at the Saltstone Disposal Facility (SDF) are forecasted activities; disposal of similar material with the associated organic contaminants have not historically occurred nor been anticipated in the SDF. To support the technical bases for disposing of these respective salt waste streams, laboratory testing and associated analyses are on-going.

Because the disposing of these waste streams with their associated organic contaminates was not anticipated at the time the current Waste Acceptance Criteria (WAC) for the Saltstone Processing Facility (SPF) and SDF was established, the WAC limits and associated Technical Safety Requirements (TSR) were not developed to support the processing of organic waste streams. A rigorous process is currently underway to determine the viability of processing these salt waste streams with organic contaminants in SPF and SDF.

The processes described in the responses to NRC Comments 6 and 37 will be followed with respect to the process formality and hierarchy of safety concerns, regulatory compliance, compliance with the performance objectives from 10 CFR 61, Subpart 61, and processibility determination of the WAC for these facilities. These processes will ensure that the appropriate limits are established for organic constituents in the waste streams and that these limits are met by the waste streams prior to authorization for these waste streams to be sent to the SPF.

As part of this extensive evaluation process, determination of the evaluation criteria for these future waste streams with respect to radionuclide leaching and waste form durability is under development. Saltstone testing for leaching and physical property characterization is identified in the Saltstone Performance Assessment (PA) Maintenance Plan. This plan is updated annually and reviewed to prioritize needs.

As an example, elements of the evaluation may include such items as:

- Continued review of available literature on organic impact on grout durability and metals/chemical leaching
- Development of testing with simulants and actual waste forms
- Characterization of the organic bio-degradation including by-products
- Characterization of the interaction between the organics, the degradation products and the grout/waste in the grout
- Understanding of the role that grout durability has on the SDF system performance with respect to radionuclide leaching.

Characteristics of the individual streams are as follows:

Tank 48 Waste

For planning purposes, project documents assume a maximum concentration of 3000 mg tetraphenylborate per liter of salt solution waste entering the SPF (Fowler 2005). This organic is known to decompose through sequential loss of the phenyl groups, eventually producing benzene. The rate of decomposition under Saltstone processing and curing conditions is currently being studied (Cozzi 2004).

To date, feasibility studies related to the disposal of Tank 48 waste Saltstone feasibility studies to date have focused on processing issues and RCRA classification (TCLP testing) of the resulting waste form. The feasibility testing completed to date or underway at the present time is summarized below.

- Processing aggregated tank waste containing the actual Tank 48 material was demonstrated and preliminary testing indicated that extraction of mercury in the TCLP leachate is not accelerated by the organics present in the actual waste

(Cozzi 2004). (Mercury is the only TCLP metal present in concentrations of concern in the waste.)

- Effects of tetraphenylborate decomposition as a function of Saltstone curing temperature and time were also investigated using a nonradioactive Tank 48 surrogate composition. Potassium tetraphenylborate was found to decompose in the Saltstone matrix in samples cured above 75 ºC (Cozzi et al. 2005). Decomposition products include benzene (Cozzi and Zamecnik 2004). An extended testing program of Saltstone made with actual Tank 48 material and simulant is being conducted to evaluate the decomposition of the tetraphenylborate during curing. To date, tetraphenylborate does not appear to affect the durability of Saltstone samples cured below 75 ºC. Consequently, an option for operating the facility which is currently being evaluated is to control the saltstone grout curing temperature to a value low enough to prevent decomposition of the potassium tetraphenylborate.

  Feasibility of controlling the pour strategy and monitoring the temperature in the vaults to accomplish this is currently being evaluated. The vaults are instrumented with thermocouples and thermal transient modeling and saltstone grout thermal property data are used to schedule the pour strategy and cell sequencing in the facility.

Actinide Removal Process

The monosodium titanate procurement specification limits the organic content of the manufactured material (<100 ppm total organic carbon; < 500 ppm alcohol -- either isopropyl or methanol) (Shah 2003). Also, most of the trace organics evaporate during storage. Since MST is added at concentrations of 0.4 g/L to the waste and subsequently filtered, the maximum potential organic contribution at SPF and SDF is very low, on the order of 0.04 ppm (Subosits 2003, p. 5). The concentration of organics in this waste stream is insignificant.

Salt Waste Processing Facility (SWPF) or Modular Caustic Side Solvent Extraction Unit (MCU)

The salt solution from the SWPF or MCU will contain entrained solvent, portions of which may transfer to the Saltstone Production Facility. This solvent consists of (0.94 wt %) a calix[4]arene-crown-6 extractant (BOBCalixC6) dissolved in an inert hydrocarbon matrix (at 69.26 wt % Isopar® L). (Delmau et al. 2002) An alkylphenoxy alcohol modifier (at 29.67 wt %) (1-(2,2,3,3-tetrafluoropropoxy)-3-(4-sec-butylphenoxy)-2-propanol, also known as Cs-7SB) added to the solvent enhances the extraction power of the calixarene and prevents the formation of a third phase. An additional additive, trioctylamine (TOA) (at 0.12 wt %), improves stripping performance and mitigates the effects of any surfactants present in the feed stream (Norato et al. 2002, p. 2).

The process designs still for both the SWPF and MCU include operations (i.e., coalescers and decanters) to recover the entrained organic. The current limit for entrained Isopar® L, the most concentrated component in the solvent and the most volatile, is still under development and, once determined, will be controlled through the WAC.

The trioctylamine is volatile and present in low concentrations in the solvent. Process handling and ventilation during the transfers before receipt to the Saltstone facility will largely evaporate this component. The Isopar® L, the major component, is a blend of alkanes similar to the solvent from PUREX processing. Its impact on saltstone properties will likely resemble those of PUREX and, therefore, the impact of the fluorinated modifier is unknown. The modifier is the least resistant of the components to chemical and radiolytic attack. The impact of the extractant is also unknown (Delmau et al. 2002, Peterson 2000).

A test program is currently being developed to perform the first phase of testing to evaluate the release of Isopar® L from Saltstone during curing and the effects of the organics carried over from salt waste decontamination processes on Saltstone leaching (Norato 2005, Cozzi 2005).

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